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ABSTRACT

This technical specialty course is one of thirty-five courses designed for use by two-year postsecondary institutions in five nuclear technician curriculum areas: (1) radiation protection technician, (2) nuclear instrumentation and control technician, (3) nuclear materials processing technician, (4) nuclear quality-assurance/quality-control technician, and (5) nuclear reactor (plant) operator trainee. This course, Nuclear Systems and Safety, contains six modules: Neutron Physics, Cross Sections and Reactions, Fission Rate and Reactor Power, Nuclear Criticality, Nuclear Material Inventory Management Techniques, and Nuclear Plant Safeguards. Each module follows a typical format including the following sections: introduction, module prerequisites, objectives, subject matter, exercises, materials, procedures, data tables, reference materials, and tests. (A master index and suggested program planning guides are also available--see CE 026 856 and CE 026 639-643.) (LRA)

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CE 026 664

Nuclear Technology Series

COURSE 24

NUCLEAR SYSTEMS AND SAFETY

CENTER FOR OCCUPATIONAL RESEARCH
AND DEVELOPMENT, INC.
4800 LAKEWOOD DRIVE
WACO, TEXAS 76710

AUG 1980

U.S. DEPARTMENT OF HEALTH, EDUCATION & WELFARE
NATIONAL INSTITUTE OF EDUCATION

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PREFACE

ABOUT NUCLEAR TECHNOLOGY MODULES

The modules – developed for use in two-year postsecondary technical institutions to prepare technicians for employment – are useful in industry for updating employees in company-sponsored training, and as laboratory texts in engineering schools. The principles, techniques and skills taught in the modules, based on tasks that nuclear technicians perform, were obtained from a nationwide advisory committee of employers of nuclear technicians. Each module was written by a working technical expert/experienced writer team and tested with students.

A module contains the following elements:

Introduction, which identifies the topic and often includes a rationale for studying the material.
Module Prerequisites, which identify the material a student should be familiar with before studying the module.
Objectives, which clearly identify what the student is expected to know for satisfactory module completion. The objectives, stated in terms of action-oriented behaviors, include such action words as operate, measure, calculate, identify and define, rather than words with many interpretations, such as know, understand, learn and appreciate.
Subject Matter, which presents the background theory and techniques supportive to the objectives of the module. Subject matter is written with the technical student in mind.
Exercises, which provide practical problems to which the student can apply this new knowledge.
Materials, which identify the equipment required to complete the laboratory procedure.
Procedures, which is the experiment section, or "hands-on" portion of the module (including step-by-step instruction) designed to reinforce student learning.
Data Tables, which are included in most modules for the first year (or basic) courses to help the student learn how to collect and organize data.
Reference Materials, which are included as suggestions for supplementary reading/viewing for the student.
Test, which measures the student's achievement of the prestated objectives.

A helpful Index has been added at the end of each course.

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The Nuclear Technology Series consists of five curricula:

Nuclear Reactor (Plant) Operator Trainee.
Radiation Protection Technician.
Nuclear Materials Processing Technician.
Nuclear Quality-Assurance/Quality-Control Technician.
Nuclear Instrumentation and Control Technician.

The completed Nuclear Technology Series consists of the following 35 technical specialty courses:

Radiation Physics.
Radiation Protection I.
Principles of Process Instrumentation.
Heat Transfer and Fluid Flow.
Introduction to Nuclear Systems.
Instrumentation and Control of Reactors and Plant Systems.
Reactor Operations.
Reactor Safety.
Reactor Auxiliary Systems.
Power Plant Systems.
Radiation Detection and Measurement.
Reactor Physics.
Power Plant Chemistry.
Introduction to Quality Assurance/Quality Control.
Metallurgy and Metals Properties.
Mechanical Component Characteristics and Specifications.
Radiation Protection II.
Radiological Emergencies.
Radiation Shielding.
Radiation Monitoring Techniques (Radiochemical).
Radioactive Materials Disposal and Management.
Advanced Radionuclide Analysis.
Nuclear Chemical Processes.
Nuclear Systems and Safety.
Radioactive Material Handling Techniques.
Nondestructive Examination (NDE) Techniques I.
Metrology.
Welding Inspection.
Civil/Structural Inspection.
Mechanical Inspection.
Quality-Assurance Practices.
Nondestructive Examination (NDE) Techniques II.
Control Systems I.
Control Systems II.
Power Plant Electrical Systems and Components.

Availability: Instructional modules are available individually or by courses from:

Center for Occupational Research and Development
4800 Lakewood Drive, Suite 5
Waco, Texas 76710
Telephone (817) 772-8756

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Printed:

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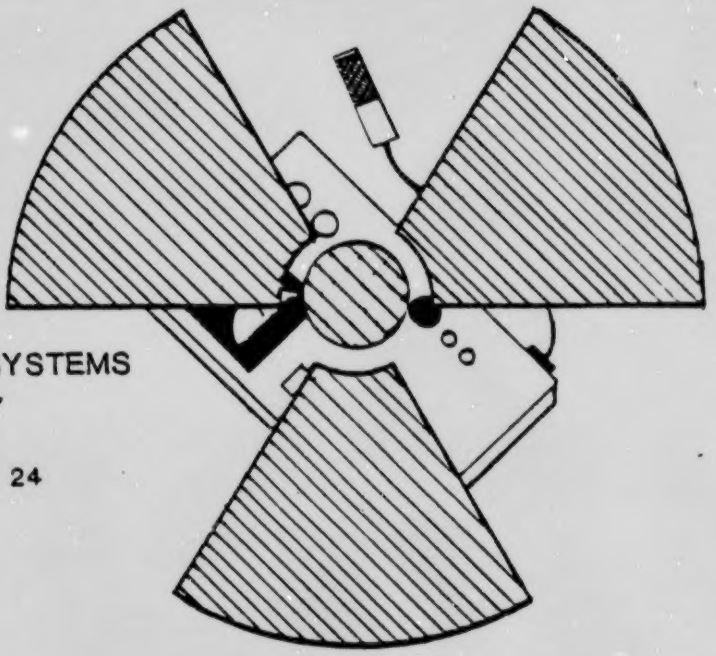
MODULE 24-6	Nuclear Plant Safeguards

NUCLEAR TECHNOLOGY

MODULE 24 - 1

NEUTRON PHYSICS



NUCLEAR SYSTEMS
AND SAFETY

COURSE 24

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COURSE MODULES

COURSE 24 NUCLEAR SYSTEMS AND SAFETY

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INTRODUCTION

The neutron is one of the basic particles composing the atom and is contained in the atomic nucleus along with the proton. Although the neutron is approximately the same size as the positively-charged proton, it possesses no electric charge. When neutrons are emitted from the atomic nucleus, they act as ionizing radiation; but since they have no charge, they do not ionize directly. Instead, they strike and interact with atomic nuclei, producing charged particles that cause ionization.

Neutrons normally are not emitted spontaneously from the nucleus, but require an external force to cause their removal. Fission is an important neutron source. This module discusses neutron sources, classification, flux and fluence, and detection.

MODULE PREREQUISITES

The student should have completed Course 1, "Radiation Physics"; Course 2, "Radiation Protection I"; Course 11, "Radiation Detection and Measurement"; Course 17, "Radiation Protection II"; and Course 23, "Nuclear Chemical Processes."

OBJECTIVES

Upon completion of this module, the student should be able to:

1. Discuss general characteristics of neutrons as presented in this module, including charge, mass, energy, range and penetrating ability.
2. List three mechanisms (nuclear reactions) by which radioactive sources can be used to release neutrons.
3. Write a nuclear equation for at least one type of radioactive neutron source.
4. Calculate neutron energy at thermal equilibrium and explain why thermal neutrons are so denoted.
5. Calculate neutron energy (given velocity or converse).
6. Calculate flux from a point source (given source strength and distance from source).
7. List and describe four categories of neutrons, based on energy.
8. List and discuss four techniques of neutron detection, giving physical principles and present applicability.
9. Define the following terms:
    - Flux.
    - Fluence.
    - Thermal neutrons.
    - Free neutrons.
    - D-T reaction.
    - D-D reaction.
    - Protron recoils.
10. Use a survey meter to detect neutrons.

The neutron is a subatomic particle. It has no charge, and its at-rest mass is approximately 1 atomic mass unit (amu) (the precise at-rest mass of the neutron is 1.008665 amu, greater by only 0.001388 amu than that of the proton). Neutron radius is about 8 x 10^{-14} cm, approximately equal to that of the proton.

Since the at-rest mass of the electron is only 0.000549 amu, unless an additional 0.000839 amu (0.7811 MeV) is somehow provided (perhaps a binding energy), it is obvious that a neutron is not simply a combination of a proton with an electron (this would violate the law of conservation of energy).

After discovery of the neutron, it was found that neutrons and protons (within a nucleus) interact constantly by transfer of other charged and neutral particles, called mesons (very light mass subatomic particles), between neutron-proton, proton-proton or neutron-neutron pairs. These so-called nucleon exchange forces are necessary to hold the nucleus together. For a variety of complex reasons, the neutron should be considered a fundamental atomic particle along with the protron and electron.

FREE NEUTRONS

Free neutrons (i.e., neutrons that exist outside a nucleus) can be produced in cosmic ray interactions. When a heavy, high-energy cosmic-ray particle (such as a proton with a kinetic energy of several hundred MeV) collides with an atomic nucleus in the upper atmosphere, the nucleus may break up into smaller units. These smaller units may be free neutrons, or protons and other nuclear fragments. Free neutrons also occur in nature, resulting from spontaneous fission of heavy elements, such as ^{244}Pu, which is found in the earth's crust.

Sometimes, a free neutron behaves like a radioactive nuclide, undergoing beta decay to become a proton according to the reaction:

$$ {}^{1}_{0}n \rightarrow {}^{1}_{1}p + {}^{0}_{-1}\beta + 0.782 \text{ MeV.} $$

Equation 24-1-1
(Equation = Eq.)

While the half-life of this reaction is 10.6 minutes, free neutrons are rapidly captured by nuclei – virtually all free neutrons are captured within a fraction of a second.

NEUTRON SOURCES

Since the number of free neutrons is very small, neutrons in quantity must be obtained from other sources for use in reactors, research laboratories or other applications. Large numbers of neutrons can be produced by appropriate nuclear reactions or from the spontaneous fission of very heavy elements. Neutron sources can be divided into two broad categories:

Radionuclide sources.
Machine sources (e.g., accelerators, reactors).

RADIONUCLIDE SOURCES

Radionuclide sources, which are radioactive, are used to produce neutrons by one of three principal processes:

(α,n) reaction.
(γ,n) reaction.
Spontaneous fission (SF).

Radioactive neutron sources are the original source of neutrons, for it was bombardment of a light nucleus (beryllium) with alpha particles that led to the discovery of the neutron:

$${}^{4}_{2}\mathrm{He} + {}^{9}_{4}\mathrm{Be} \rightarrow {}^{12}_{6}\mathrm{C} + {}^{1}_{0}\mathrm{n} + Q\ (\mathrm{MeV}). \qquad \text{Eq. 24-1-2}$$

This (α,n) reaction is one of two commonly used to produce neutrons. In the (α,n) reaction, an alpha emitter (such as radium or plutonium) is mixed with a low-Z (atomic number) element (such as beryllium or boron). The resulting (α,n) reaction (an α in, an n out) produces the energetic neutrons of energy Q (MeV).

Neutron yield depends on many factors, including:

Energy of the incident alpha particles.
Low-Z material being irradiated.
Completeness of mixture of the two ingredients.
Self-absorption of alpha particles and neutrons.
Source geometry.
Impurity content.

The other principal nuclear reaction commonly used to produce neutrons is the (γ,n), or photoneutron reaction. In this reaction, photoneutrons are produced by mixing nuclide-emitting, suitably high-energy gamma rays with a low-Z element (e.g., beryllium or deuterium).

Only beryllium or deuterium can produce neutrons with gamma-ray energies below 5 MeV. For beryllium the photoneutron production threshold is 1.67 MeV (for deuterium it is 2.23 MeV); thus, a source must produce gamma rays with energies above 1.7 MeV if beryllium is used.

Spontaneous fission (SF) sources are heavy elements, generally all artifically produced, that undergo fission as a mode of radioactive decay. The fission is accompanied by emission of two or three neutrons and several gamma rays. SF sources are radioactive, in the common sense, and also undergo alpha decay; thus, two competitive decay modes, each has its own characteristic half-life. The alpha decay mode has a greater probability of decay, hence a shorter half-life (Table 24-1-1).

MACHINE SOURCES

Machine sources of neutrons include reactors and particle accelerators. In general, these sources are not widely employed because of their sizes. A possible exception is the so-called D-T (deuterium-tritium) generator, a small positive-ion accelerator that utilizes the D-T reaction for generating neutrons by bombarding a tritium-rich target ($^{3}_{1}H$) with positive deuterium ions ($^{2}_{1}H$):

$$^{2}_{1}H + ^{3}_{1}H \rightarrow ^{4}_{2}He + ^{1}_{0}n + Q\ (\text{MeV}). \qquad \text{Eq. 24-1-3}$$

Q = 17.6 MeV; all generated neutrons have the same energy, ~ 14 MeV; the other 3.6 MeV is associated with the liberated α particle.

Another neutron-generating machine sometimes used is a D-D (deuterium-deuterium) generator; it utilizes a D-D reaction. This machine is analogous to the D-T generator except that the target is deuterium rich rather than tritium rich. Some commercial units are made with interchangeable targets so that neutrons can be generated by either a D-D or a D-T reaction.

TABLE 24-1-1
COMMON NEUTRON SOURCES AND THEIR PROPERTIES

SOURCE	REACTION	HALF-LIFE	MEAN NEUTRON ENERGY (MeV)	APPROX. YIELD (n/sec·Ci)	COMMENTS
^{124}SbBe	(γ,n)	60 days	0.024	1.6×10^{6}	Short half-life renders this source unsuitable for many applications; has high-intensity gamma field.
^{210}PoB	(α,n)	138 days	2.3	6.0×10^{5}	
^{210}PoBe	(α,n)	138 days	4.5	2.5×10^{6}	
^{210}PoLi	(α,n)	138 days	0.5	6.1×10^{4}	Yield relatively low.
^{226}RaBe	$\left\{\begin{matrix}(\alpha,n)\\(\gamma,n)\end{matrix}\right\}$	1640 years 1640 years	3.9 0.3	$\left\{\begin{matrix}1.5 \times 10^{7}\\1.2 \times 10^{6}\end{matrix}\right\}$	High-intensity gamma field.
^{227}AcBe	(α,n)	21.8 years	4.6	$\sim 1.2 \times 10^{6}$	Not generally available.
^{228}ThBe	(α,n)	1.9 years	0.8	$\sim 2.5 \times 10^{6}$	Not generally available; high-intensity gamma field.
^{238}PuBe	(α,n)	87.8 years	4.5	1.8×10^{6}	Common; used for calibration of instruments.
^{239}PuBe	$\left\{\begin{matrix}(\alpha,n)\\(SF)^{+}\end{matrix}\right\}$	$\left\{\begin{matrix}\text{24,360 years}\\\text{24,360 years}\end{matrix}\right\}$	4.5 1.2¶	$\left\{\begin{matrix}1.5 \times 10^{6}\\0.5\end{matrix}\right\}$	Common; used for calibration of instruments.
^{239}PuF$_4$	(α,n)	24,360 years	0.7	$\sim 10^{6}$	Physically relatively large, much self-absorption.
^{241}AmBe	(α,n) (SF)$^{+}$	458 years	4.4 1.2¶	2.2×10^{6} 1.9	Becoming common; good reactor starter source.
^{240}Pu	SF^{+}	6600 years	1.2¶	3500*	Relatively weak for practical use.
^{244}Cm	SF^{+}	18.4 years	1.2¶	1.5×10^{5}*	High gamma field.
^{252}Cf	SF^{+}	2.65 years	1.2¶	4.3×10^{9}*	Coming into wide use as calibration and starter source.
^{249}Bk	SF^{+}			3.6×10^{5}*	Not yet available in quantity.

$^{+}$ SF = spontaneous fission.

¶ Mean energy of fission neutron spectrum.

* Yield is given in n/sec-g.

Accelerators of various kinds – betatrons, cyclotrons, linacs, etc., – can be used to produce neutrons from a wide variety of nuclear reactions. In addition to the D-D and D-T reactions, the (p,n) reaction is also widely used in accelerators. In this reaction, a proton is accelerated into a target nucleus to produce neutrons:

$$ {}^{1}_{1}H + {}^{A}_{Z}X \rightarrow {}^{A}_{(Z+1)}Y + {}^{1}_{0}n + Q. \qquad \text{Eq. 24-1-4} $$

where

X = target nucleus.

Y = product after reaction.

[In general, fission-produced neutrons (from reactors) are used for research, educational and analytical purposes as well as for power production; the remaining modules in this course provide a more comprehensive consideration of this area.]

Neutron sources can produce neutrons over a wide range of energies (Figure 24-1-1). As can be seen (Table 24-1-1), Ra-Be is primarily an (α,n) source. The spectrum shows peaks at about 1 MeV and 4 MeV, and a continuous (i.e., neutrons of all energies) spectrum from 0 to 12 MeV.

Most (α,n) reactions produce a neutron energy spectrum with a peak in the region of 3 to 5 MeV. Photoneutron sources also may produce continuous-energy spectra, or they may produce neutrons with only one or a few characteristic energies. The relatively common ^{124}Sb-Be source, usually made by surrounding a rod of ^{124}Sb with a sleeve or cladding of beryllium, emits neutrons in one single energy group: approximately 0.03 MeV.

Other neutron-producing nuclear reactions, such as the D-T, results in monoenergetic (single-energy) neutrons, or a very narrow spectrum of energies. As was mentioned above, the D-T reaction produces neutrons with energies of 14 MeV, while the D-D reaction produces neutrons with energies primarily in the few-MeV region.

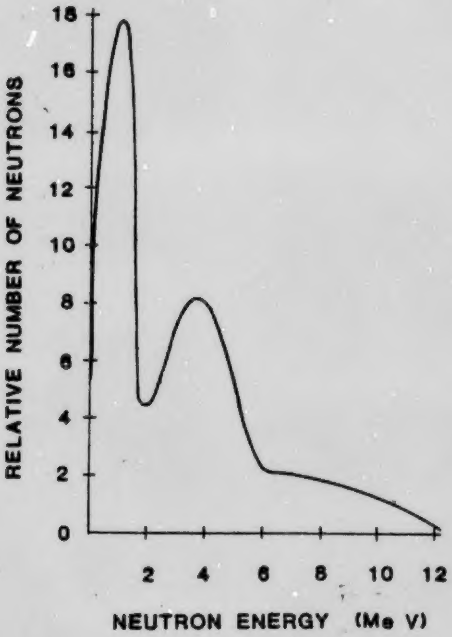


Figure 24-1-1.
Neutron energy spectrum from Ra-Be (α,n) source.

A fission neutron spectrum (Figure 24-1-2) is significantly different from spectra produced by other nuclear reactions (Figure 24-1-1). Fission neutrons are produced over a broad range of energies, from 0 to about 18 MeV. The number of neutrons in a given energy group is roughly inversely proportional to the log of the energy over the range from about 1 to 18 MeV. In this range, probability of fission goes up as incident neutron goes down, because

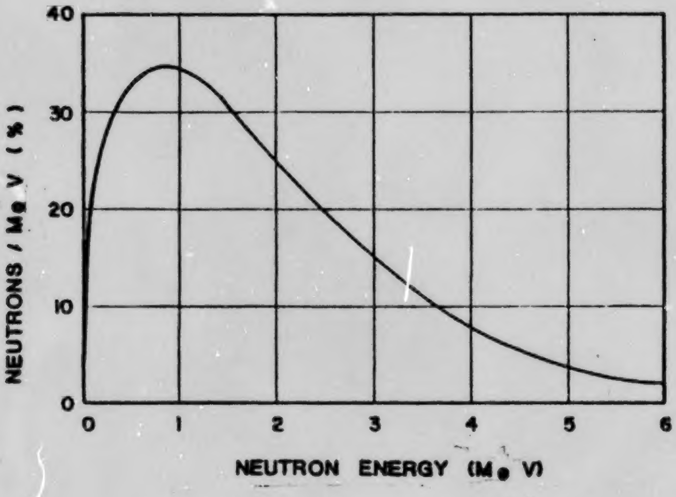


Figure 24-1-2. Energy distribution of neutrons from thermal fission of ^{235}U.

each neutron has a greater chance of interaction with a nucleus if it is moving slowly. (It spends more time in the vicinity of a target nucleus!) A peak in neutron production rate occurs at about 0.7 MeV. The decrease below 0.7 MeV is due to self-absorption in the source.

It is instructive to characterize neutron spectrum in terms of its average neutron energy:

$$\bar{E} = \frac{\Sigma_o^i n_i E_i}{N} \qquad \text{Eq. 24-1-5}$$

where

$\bar{E}$ = average neutron energy.

i = any specific group of neutrons with energy E_i, or between E_1 and E_2 centered about E_i.

n_i = number of neutrons with energy E_i, or with energies between E_1 and E_2 centered about E_i.

N = total number of neutrons.

Σ_o^i = sum of products of N_i and E_i over all energy groups (from group 0 to group i).

CLASSIFICATION OF NEUTRONS

Neutrons can be characterized in many different ways. One is by their method of production, as with the neutron sources. More commonly, however, neutrons are categorized according to their energy; the type of nuclear reaction or interaction with matter is strongly dependent upon neutron energy.

Neutron energies generally range from about 10^{-4} eV to about 1000 MeV – more than 13 orders of magnitude.

Commonly, neutrons are put into four broad energy categories:

Thermal.
Intermediate.
Fast.
Relativistic.

Thermal neutrons are neutrons in thermal equilibrium with their surroundings. This statement means that neutron energy is about the same as the energy of surrounding atoms that are in constant vibrational motion. This motion depends on temperature, hence the term "thermal neutrons."

Actually, a whole distribution of neutron energies exists in this situation (Figure 24-1-3) – known as the Maxwellian, or Maxwell-Boltzmann, distribution. The most probable energy of a thermal neutron can be calculated from kinetic theory:

$$E = KT \qquad \text{Eq. 24-1-6}$$

where

E = energy of neutron, ergs.

K = Boltzman constant

$= 1.38 \times 10^{-16}$ erg/°K.

T = temperature in °K (absolute temperature).

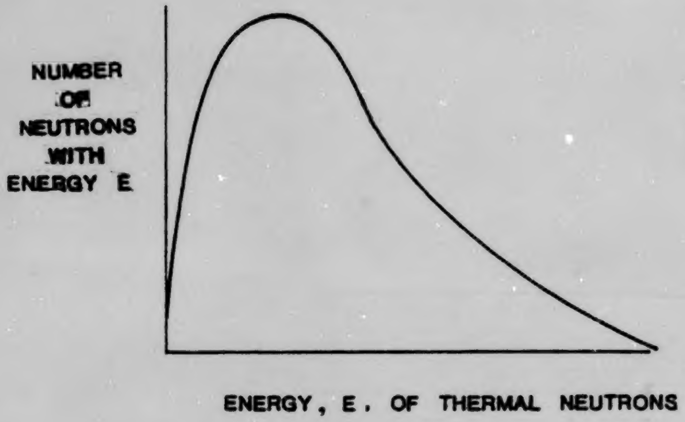


Figure 24-1-3. Maxwell-Boltzmann distribution of neutrons in thermal equilibrium. Peak occurs at 0.025 eV for room temperature.

EXAMPLE

At room temperature (68°F ~ 20°C ~ 239°K), the most probable energy is easily calculated as follows:

$$E = 1.38 \times 10^{-16}\ \text{erg/°K} \times 293\text{°K}$$
$$= 4.04 \times 10^{-14}\ \text{ergs}$$

or, because there are 6.24×10^{11} eV/erg,

$$E = 6.24 \times 10^{11}\ \text{eV/erg} \times 4.04 \times 10^{-14}\ \text{ergs}$$
$$= 0.025\ \text{eV at } 20\text{°C}.$$

Thermal neutrons generally are considered to be monoenergetic at this most probable energy, and this simplifying assumption provides sufficient accuracy for most purposes.

Thermal neutrons, for practical purposes, are taken to be neutrons with an energy of 0.025 eV.

Because neutron energy is actually kinetic energy, it is possible to relate the velocity of a neutron to its energy by the following relationship from classical physics:

$$KE = 1/2\ mv^2 \qquad \text{Eq. 24-1-7}$$

where

KE = kinetic energy (energy of motion) of neutron, ergs.
m = neutron mass, grams.
v = neutron velocity, cm/s.

Thus, one erg is a gram cm^2/s^2.

For a 20°C thermal neutron, KE = 0.025 eV = 4.04×10^{-14} erg.

EXAMPLE

Velocity of a thermal neutron may be calculated as follows:

$$4.04 \times 10^{-14} \text{ ergs}$$
$$= 1/2 \times 1.67 \times 10^{-24} \text{ grams} \times v^2$$
$$v^2 = 4.84 \times 10^{12}\ cm^2/s^2 \text{ or ergs/grams}$$
$$v = 2.2 \times 10^6 \text{ cm/s} = 2200 \text{ m/s.}$$

Thus, the velocity of a thermal neutron, in equilibrium at 20°C (room temperature), is 2200 m/s, or about 5000 miles per hour.

Even though thermal neutrons are taken, for practical purposes, to have energies of 0.025 eV, they actually range up to about 0.5 eV. The upper end of the thermal region (from 0.025 eV to 0.5 eV) sometimes is referred to as the epithermal region, and neutrons with energies in this range are called "epithermal neutrons."

"Intermediate neutrons" are those with energies in the region 0.5 eV to 10 keV.

"Fast neutrons" are those with energies between 10 keV and 1 MeV. From a radiation protection standpoint, these are the most hazardous neutrons. The velocity of fast neutrons can be calculated in the manner used for thermal neutrons.

"Relativistic neutrons" are those with energies greater than 10 MeV. This energy, which generally is greater than the binding energy of most nucleons, is particularly effective at fragmenting nuclei. The term relativistic comes from Einsteinian physics, which notes that, as the velocity of an object approaches the velocity of light, it gets heavier and heavier until, at the velocity of light, its mass becomes infinite.

The four energy groups are most useful in characterizing types of nuclear interactions that predominate in each energy region. Thermal (also known as slow) neutrons are most commonly involved in nuclear reactions that result in their capture by a nucleus, which includes emission of gamma ray, and most fission reactions.

The absorption or capture of intermediate energy neutrons is marked by a series of peaks or points of high affinity for neutrons of a particular energy. These are known as resonance peaks; hence intermediate neutrons, sometimes known as resonance neutrons, are target specific. This feature is of great utility in analysis of materials. Fast neutrons exhibit a variety of nuclear reactions, but capture reactions do not necessarily predominate, although under some conditions they may. Examples of other reactions are (n,α), (n,p) and $(n,2n)$.

Very high-energy or relativistic neutrons have the ability to produce complex nuclear reactions, including "spallation" (a reaction in which a target nucleus is broken up into several fragments).

There are other neutron classifications. Often, it is convenient to use two broad categories: fast and slow (thermal). Neither has an exact or precise definition. Other, more specific, characterizations are discussed in subsequent modules in this Course.

NEUTRON FLUX AND FLUENCE

Neutrons are most commonly and universally measured in terms of neutron flux, more commonly referred to as flux (ϕ). Flux is the number of neutrons passing through a unit area (1 cm^2) in a unit of time (1 s). Thus, neutron flux, a rate of neutron flow across a boundary (e.g., detection aperture), stated as neutrons per square centimeter per second (n/cm^2-s or $n/cm^2/s$), is independent of energy.

It is, of course, possible to qualify flux by use of an appropriate descriptor, such as "thermal" or "fast." Thus, one can quite properly and precisely talk of thermal flux, ϕth, the number of thermal neutrons (Nth) per cm^2 per second.

Neutron flux can be calculated from sources of known strength and geometry by various mathematical expressions (most radionuclide neutron sources and accelerator targets are sufficiently small to be considered as point sources):

$$\phi = \frac{S}{4\pi d^2} \qquad \text{Eq. 24-1-8}$$

where

S = source strength (or yield), neutrons/s.

d = distance from source.

(Note: This calculation ignores absorption and scattering.)

Neutron flux sometimes is represented by the letters NV:

$$\phi = NV.$$

The quantity NV is simply a product of the number of neutrons/unit volume in (n/cm^3) and their velocity in (cm/s), which dimensionally has the appropriate units of flux, viz n/cm^2-s. The NV expression often is found in analysis and reactor physics work.

If flux (or average neutron flux) is multiplied by time or irradiation (or exposure), the total number of neutrons passing through a unit area (1 cm^2) is obtained. This quantity, known as fluence, is expressed in units of n/cm^2. (Fluence is symbolized by the Greek letter ψ or by NVt.) In the NVt expression, t refers to time, and the unit is again dimensionally n/cm^2. Note that the fluence rate is merely flux and has the appropriate units. Occasionally, fluence is referred to as the "time integral of flux" or flux time integral. This expression is correct, having been derived from calculus notation, but is cumbersome. Hence, the simple term fluence is preferred.

Neutrons also can be measured in other units. The familiar unit of absorbed dose – the rad – can be used to quantify neutron dose, and the biological dose from neutrons can be expressed in units of rem (Roentgen equivalent man).

NEUTRON DETECTION

Because neutrons are uncharged, they cannot be directly detected by ordinary electronic means; however, because of their ability to cause nuclear reactions of various types, neutrons can be detected and measured by several indirect methods, which include the following:

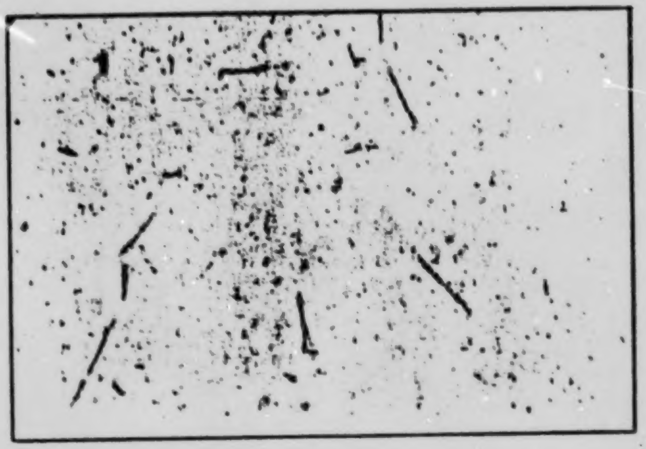
Nuclear emulsions.
Fission counters.
BF_3 proportional counters.
Activation analysis.

Generally, these neutron-measuring techniques require some means of defining the spectrum to obtain absolute flux measurements. Otherwise, only relative values of flux in a fixed spectrum are obtained.

NUCLEAR EMULSIONS

Nuclear emulsions are essentially thick pieces of photographic material that contain a high percentage of hydrogen. As neutrons pass through the emulsion, they knock some of the hydrogen atoms loose and give them energy. These charged particles, known as proton recoils, move a small distance through the photographic emulsion. When the film is developed and examined under a microscope, a tiny track is seen that marks travel of the proton recoil (Figure 24-1-4). For a given neutron spectrum, the number of tracks per unit area is proportional to the neutron fluence.

Figure 24-1-4.
Proton recoil tracks in
nuclear emulsion.



Nuclear emulsions are widely used as personnel monitoring devices.

(Note: It should be noted that nuclear emulsions are not very good for measuring neutron exposure because of variation of quality factor with energy, unless the spectrum is well characterized. Problems also arise because emulsions are insensitive to important parts of the neutron energy spectrum.)

FISSION COUNTERS

A fission counter is a device that utilizes uranium or some other fissionable material. When struck by neutrons of an appropriate energy, fission occurs and charged particles and gamma rays are produced. If the fissionable material has been used to line a counter tube (e.g., a Geiger-Mueller tube or proportional counter tube) a direct electronic readout of the fission events can be made, which provides a measurement of flux (the number of fissions per second is directly proportional to the neutron flux). A fission chamber provides a measurement of flux if the spectrum is known and the counter is properly calibrated.

BF_3 PROPORTIONAL COUNTERS

Perhaps the most widely used neutron flux detector is the BF_3 (boron trifluoride) proportional counter. This device works in the same fashion as a conventional proportional counter, except that, in this case, the proportional counter tube is filled with BF_3 gas. When placed in a neutron flux, the following nuclear reaction occurs:

$$^{10}_{5}B + ^{1}_{0}n \rightarrow ^{4}_{2}He + ^{7}_{3}Li + Q. \qquad \text{Eq. 24-1-9}$$

Alpha particles, produced inside the detector, are appropriately detected. A BF_3 proportional counter makes an excellent flux detector because it is especially sensitive to thermal neutrons.

ACTIVATION ANALYSIS

Activation analysis is a useful technique for monitoring fluence. Appropriate elements, placed in the neutron field, capture neutrons and become radioactive:

$$^{197}_{79}Au + ^{1}_{0}n \rightarrow ^{198}_{79}Au + Q. \qquad \text{Eq. 24-1-10}$$

The radioactive ^{198}Au emits a beta particle with a half-life of 2.7 days. Thus, by determining the amount of radioactivity and making appropriate calculations, neutron fluence can be determined.

There are, of course, many other neutron detectors, including neutron scintillators, boron-lined chambers of various types, and a variety of β-γ detectors loaded with materials that become activated. Also used are boron-loaded films and thermoluminescent materials, the latter being of particular value in personnel monitoring work.

NEUTRON SURVEY METERS

Virtually all neutron survey meters use BF_3-filled gaseous detector tubes and function as proportional counters. BF_3 tubes may be surrounded by paraffin, polyethylene or some other hydrogenous material, which acts to slow down and thermalize fast neutrons. This feature makes the instrument more sensitive because the $^{10}B(n,\alpha)\ ^{7}Li$ reaction is more likely to occur with thermal or slow neutrons than with fast neutrons.

Neutron survey meters can be used to gain information on neutron flux levels. Their readings must be carefully considered and evaluated because response of the instrument is highly dependent on neutron energy. Depending upon the amount of hydrogenous material surrounding the BF_3 tube, sensitivity can be several hundredfold greater for thermal neutrons that for fast neutrons; thus, in the same flux of a ^{226}Ra-Be source, which is mostly fast neutrons, and a fission source as moderated in a reactor, enormously different readings might be obtained. Neutron instruments should be appropriately calibrated, therefore, by using the same type of neutron spectrum as is to be measured. In particular, this calibration is important if flux readings are to be converted to dose rate (errors of several hundredfold are possible and could have serious consequences).

A technique sometimes used to crudely estimate thermal and fast flux is to compare the reading of a bare BF^3 proportional detector with one wrapped with cadmium. The cadmium absorbs virtually all the thermal neutrons, and, by noting the ratio of the bare-to-shielded readings, a crude idea of the spectrum can be obtained. Such techniques should be carried out by experienced individuals and should not supplant more accurate methods.

For personnel dose rate measurements, a BF^3 detector can be wrapped in a specially designed polyethylene-boron sleeve, which makes its response to neutrons approximate that from a human body. Such an instrument can be directly calibrated in units of rem/hr, or rad/hr.

EXERCISES

1. Use Table 24-1-1; calculate neutron flux expected at one foot (30.48 cm) from a 1.5-Ci ^{238}PuBe source.
2. Calculate the most probable energy of a thermal neutron in a reactor in which the temperature is 412°F; calculate its velocity.
3. Calculate velocity of a thermal neutron on a hot (110°F) summer day; on a cold (-10°F) winter day.
4. Write an equation for the (γ,n) reaction on beryllium.
5. Write a nuclear equation for the (D-D) reaction.

MATERIALS

1 neutron source, PuBe, 5 curies (80 grams).
1 neutron survey meter.
2 x 3 cycle log-log paper.

PROCEDURES

1. Use survey meter; measure flux or dose rate from a neutron source at these distances: 15 cm, 30 cm, 100 cm, 200 cm; plot flux versus distance on log-log paper; compare with calculated values based on source strength; discuss reasons for differences noted.
2. Repeat the above procedure; use a detector wrapped with cadmium. How do results compare with those above? Explain differences.

DATA TABLE

None.

REFERENCE MATERIALS

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TEST

1. Describe a typical neutron; use such characteristics as charge, mass, energy range and penetrating ability:

2. List three nuclear reactions by which radioactive sources can be made to emit neutrons:

3. Write a nuclear equation associated with each of the following types of neutron sources:

   α,n.

   γ,n.

   Spontaneous fission.

4. Calculate energy of a thermal neutron at a temperature of 25°C:

5. Calculate kinetic energy of a neutron with a velocity of 5×10^6 cm/s:

6. Calculate flux from a 5-curie ^{238}PuBe at a distance of 10 feet:

7. List four neutron energy categories; include energy range associated with each:

8. List four techniques of neutron detection; include principles of each:

9. Define the following terms:
   a. Thermal neutron.

   b. Free neutron.

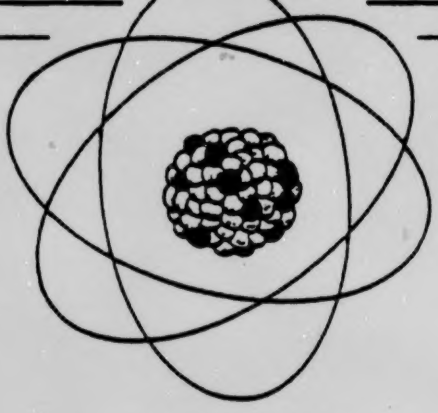
   c. Flux.

   d. Fluence.

   e. D-T reaction.

   f. D-D reaction.

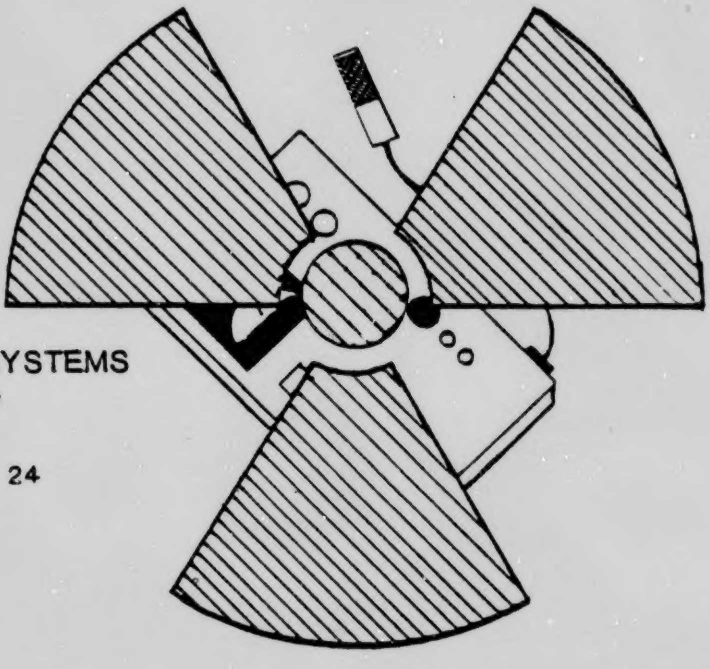
   g. Proton recoils.



NUCLEAR TECHNOLOGY

MODULE 24 - 2

CROSS SECTIONS AND REACTIONS



NUCLEAR SYSTEMS
AND SAFETY

COURSE 24

ORD CENTER FOR OCCUPATIONAL RESEARCH AND DEVELOPMENT

COURSE MODULES

COURSE 24 NUCLEAR SYSTEMS AND SAFETY

MODULE 24-1	Neutron Physics
⇨ MODULE 24-2	Cross Sections and Reactions
MODULE 24-3	Fission Rate and Reactor Power
MODULE 24-4	Nuclear Criticality (0530 ERDA Manual)
MODULE 24-5	Nuclear Material Inventory Management Techniques
MODULE 24-6	Nuclear Plant Safeguards

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INTRODUCTION

This module describes several neutron interactions with matter. Possible interactions and their probabilities (as well as other concepts) should be understood before the fission process is studied. "Cross section" gives an indication of the probability of a particular type of interaction taking place. Its application and methods of measurement are introduced, along with basic laboratory procedures demonstrating absorption, attenuation, thermalization and activation.

MODULE PREREQUISITES

The student should have completed Course 1, "Radiation Physics"; Course 2, "Radiation Protection I"; Course 11, "Radiation Detection and Measurement"; Course 17, "Radiation Protection II"; and Course 23, "Nuclear Chemical Processes."

OBJECTIVES

Upon completion of this module, the student should be able to:

1. List three different types of reactions that produce neutrons.
2. Write typical reaction equations for radioactive capture and charged-particle emission.

3. Describe processes that take place during these neutron scattering reactions:
    - Head-on collision with a small nucleus.
    - Glancing collision with a small nucleus.
    - Head-on collision with a large nucleus.
    - Glancing collision with a large nucleus.
4. Describe cross-section concept; state why it is not equal to cross-sectional area of a nucleus.
5. Use "Chart of the Nuclides" and Radiological Health Handbook to research scattering, absorption and fission cross sections.
6. List differences between microscopic and macroscopic cross sections.
7. Calculate macroscopic cross section, given only the radioisotope and its neutron density.
8. Draw a typical absorption cross section versus energy curve; explain significance of three major parts of curve.
9. Explain reason for reaction rate per cm^3 to be given as $\Sigma\phi$ and total reaction rate by $\Sigma\phi V$.
10. Define the following terms:
    - Nuclear reaction.
    - Elastic scattering.
    - Inelastic scattering.
    - Capture.
    - Compound nucleus.
    - Fission.
    - Cross section.
    - Microscopic cross section.
    - $\frac{1}{v}$.
    - Resonance.
    - Macroscopic cross section.

37

Albedo.
Cadmium ratio.
Neutron age.
Neutron reaction rate.

11. Experimentally determine macroscopic and microscopic absorption cross sections of a known material by using a portable neutron survey meter and a monoenergetic beam of neutrons.
12. Experimentally determine Cd ratio for a given nuetron flux.

ELEMENTARY NUCLEAR REACTIONS

Nuclear reactions are those that occur within the nucleus of an atom. They usually are governed by internal forces, which can result in a radioactive nucleus, or by external forces, such as a bombardment by an energetic particle. Nuclear reactions may result in a variety of changes, including production of a nucleus of different elements or production of radiation.

ENERGETICS OF NUCLEAR REACTIONS

The general equation characterizing a nuclear reaction can be written as:

$$A + B \rightarrow C + D + Q. \qquad \text{Equation 24-2-1}$$

(Equation = Eq.)

In the equation, A, B, C and D are various particles (or nuclei), and Q refers to the necessary mass balance or energy term. An example of a nuclear reaction, taken from Module 12-1, "Neutron Physics," is the production of neutrons by alpha bombardment of beryllium nuclei:

$${}^{4}_{2}\mathrm{He} + {}^{9}_{4}\mathrm{Be} \rightarrow {}^{1}_{0}\mathrm{n} + {}^{12}_{6}\mathrm{C} + Q. \qquad \text{Eq. 24-2-2}$$

In this equation, not only is the reaction shown, but certain notations have been added to aid in illustrating basic rules of such reactions. These notations are atomic number, Z, and mass (or nucleon) number, A. The number of nucleons (protons plus neutrons) on the left side of equation (4 + 9) exactly equals the number of the right side (1 + 12). Similarly, the atomic number [i.e., the number of nuclear charges on the left side (2 + 4) equals the number on the right side (0 + 6)]. Two of the three conservation rules governing nuclear reactions are illustrated:

> In any nuclear reaction, the mass (nucleon) number is conserved.
>
> In any nuclear reaction, the atomic number is conserved.

The third conservation rule:

> In any nuclear reaction, energy is conserved.

This rule is incorporated in the Q term. If each reactant in the equation is carefully weighed, masses of those on the left side do not equal those on the right. The slight difference is shown by detailed calculation:

Rest mass of a ${}^{4}_{2}\text{He}$ atom = 4.002603 amu
Rest mass of a ${}^{9}_{4}\text{Be}$ atom = $\underline{9.012186}$ amu
Total mass, left side = 13.014789 amu
Rest mass of a ${}^{1}_{0}\text{n}$ neutron = 1.008665 amu
Rest mass of a ${}^{12}_{6}\text{C}$ atom = $\underline{12.000000}$ amu
Total mass, right side = 13.008665 amu

Mass difference = 13.014789 - 13.008665
= 0.006124 amu.

The mass difference (the value of Q in the equation) is known as "mass defect."

Because matter can be neither created nor destroyed, the missing mass must be accounted for. If $E = mc^2$ is used for accounting, the missing mass has been converted to energy and is actually the kinetic energy radiated or imparted to the products on the right side. The kinetic energy imparted to each particle is in inverse proportion to the mass of the products; thus, each neutron gets about 12 times as much of the kinetic energy (about 12/13 of the total) as does each carbon nucleus.

The apparent conversion of mass into energy in a nuclear reaction is explainable in terms of the binding energy needed to hold various nuclei together. In general, the amount of energy required to hold a large nucleus together is greater than that required to hold a smaller nucleus together. However, the average energy per nucleon may be less. The least amount of energy per nucleon is required by intermediate nuclei, such as iron and copper, while very light nuclei require considerably more binding energy per nucleon (Figure 24-2-1).

The nuclear reaction can be written in short form. The example reaction equation, given earlier, could be expressed as $^{9}\mathrm{Be}\ (\alpha,n)^{12}\mathrm{C}$, or merely as the (α,n) for $^{9}\mathrm{Be}$. The former is preferred.

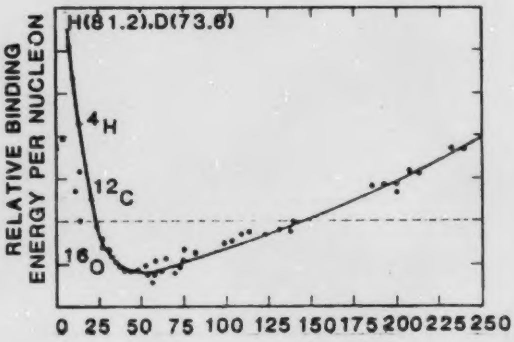


Figure 24-2-1. Average energy per nucleon.

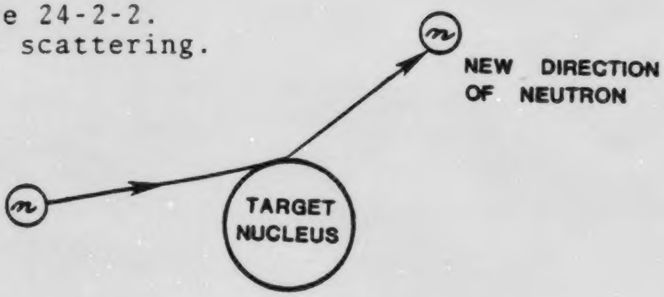
TYPES OF NUCLEAR REACTIONS

Five basic nuclear reactions can occur with neutrons as the incident or "bombarding" particle:

Elastic scattering.
Inelastic scattering.
Neutron capture.
Particle ejection.
Fission.

Elastic scattering is a reaction in which kinetic energy is conserved. It is sometimes given the name "billiard-ball collison" because the neutron may be thought of as bouncing off the target nucleus, much in the same manner as one billiard ball bounces off another. The neutron need not strike the target nucleus, but may be deflected by its nuclear field (Figure 24-2-2).

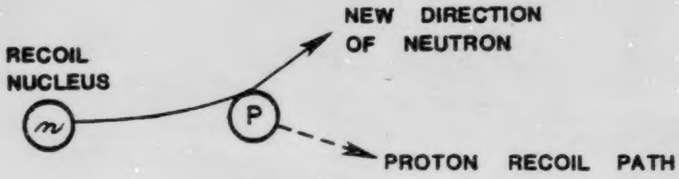
Figure 24-2-2.
Elastic scattering.



Elastic scattering is more apt to occur with fast neutrons than with slow ones. If the target nucleus is of the same size as a neutron (such as a hydrogen nucleus), it may be given enough kinetic energy to move (or be knocked out of) a chemical compound. This reaction is known as recoil nucleus (Figure 24-2-3). Proton recoil is an important method of fast neutron detection (discussed in the previous module).

Elastic scattering reactions sometimes are known as (n,n') reactions, signifying neutron in (n) at a given velocity and a neutron out at a different velocity (n').

Figure 24-2-3.
Recoil nucleus.


NEW DIRECTION
OF NEUTRON
RECOIL
NUCLEUS
n
P
PROTON RECOIL PATH

Inelastic scattering is a process in which some kinetic energy is lost. The incident neutron actually may enter and pass through the target nucleus, transferring some of its kinetic energy to the target nucleus. It also may accomplish this by passing very close (i.e., through the nuclear field) to the target nucleus. The energy imparted to the target nucleus puts it in a higher energy or excited state. When the target nucleus returns to the ground state, it emits this excess energy in the form of one or more gamma rays (Figure 24-2-4).

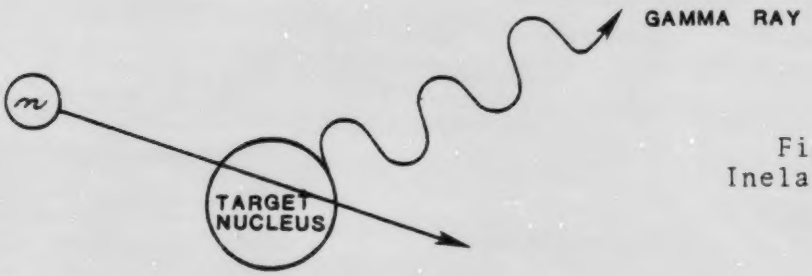

GAMMA RAY
n
TARGET
NUCLEUS

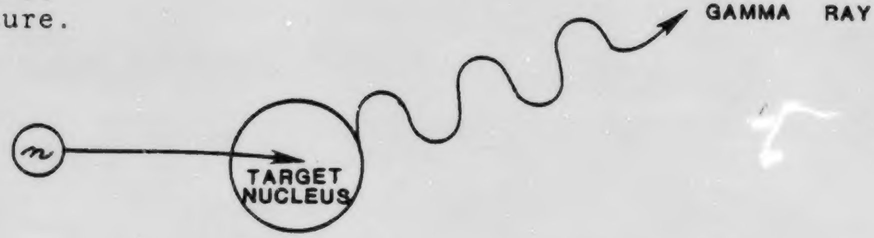
Figure 24-2-4.
Inelastic scattering.

The gamma ray usually is emitted instantaneously after the reaction, although some excited nuclei having relatively long half-lives may be produced. Inelastic scattering reactions of the type described are also (n,n') reactions and occur most often with intermediate and slow neutrons. It should be noted that, in both inelastic and elastic scattering, the incident particles (neutron and target nucleus) are the same as the product nuclei. There are no changes in the reactants except energy transfers.

Capture occurs when a neutron enters a target nucleus and remains there (Figure 24-2-5), producing a new nucleus with a mass number increased by 1:

$$ {}^{1}_{0}n + {}^{59}_{27}Co \rightarrow {}^{60}_{27}Co + Q. \qquad \text{Eq. 24-2-3} $$

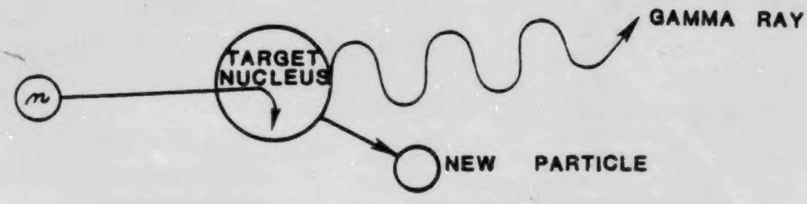
Figure 24-2-5.
Neutron capture.



The new nucleus, which is in an excited state, emits this excess energy as a gamma ray when it returns to ground state. Thus, capture reaction is sometimes called (n,γ) reaction. The new nucleus also may be radioactive, as in the example (Equation 24-2-3). Capture reactions are more apt to occur with thermal or slow neutrons than with fast ones.

Particle ejection occurs when a neutron enters a target nucleus and, because energy is imparted to the new nucleus, results in the emission of another particle (Figure 24-2-6). The nucleus remaining after emission of the particle may be in an excited state, returning to the ground state by the emission of gamma radiation. The remaining nucleus also may be radioactive. A typical neutron reaction that produces particle ejection is shown by Equation 24-2-4.

Figure 24-2-6.
Particle ejection.



$$ {}^{1}_{0}n + {}^{10}_{5}B \rightarrow {}^{4}_{2}He + {}^{7}_{3}Li + Q. \qquad \text{Eq. 24-2-4} $$

In this reaction a neutron bombards a boron-10 nucleus, resulting in the emission of an alpha particle, and producing a lithium-7 nucleus (Figure 24-2-7). A compound nucleus, formed as an intermediate stage, quickly decays to an alpha particle and a lithium-7 nucleus. Because it is so unstable, the compound nucleus may have a lifetime of microseconds or less.

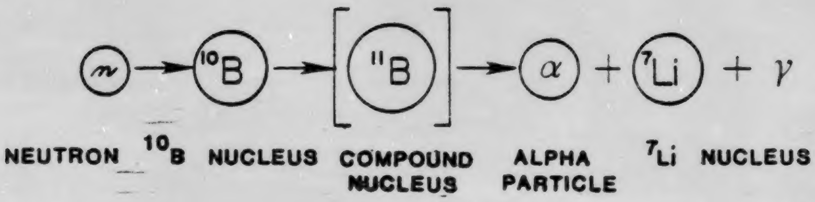

n
10B
11B
α
7Li
γ
NEUTRON
10B NUCLEUS
COMPOUND NUCLEUS
ALPHA PARTICLE
7Li NUCLEUS

Figure 24-2-7. Particle ejection reaction, showing formation of compound intermediate nucleus.

Neutron fission occurs when a neutron causes a target nucleus to break into two nuclei, with the probable emission of additional neutrons and gamma rays (Figure 24-2-8). A neutron interacts with a ^{235}U nucleus to form a compound ^{236}U nucleus, which almost instantaneously splits into two smaller (but still fairly large) nuclei (such as ^{131}I and ^{102}Mo), emitting three neutrons and gamma rays in the process. Both fission products are radioactive. (Fission is studied in detail in following modules.)

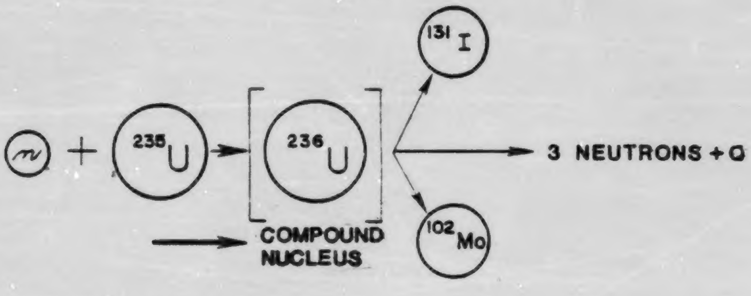

n
235U
236U
131I
102Mo
3 NEUTRONS + Q
COMPOUND NUCLEUS

Figure 24-2-8. Typical fission reaction.

Other neutron reactions (in addition to the basic five) include spallation. This reaction, which occurs with very high-energy neutrons, is the breakup of a relatively light nucleus into small particles. For example, a ^{7}Li nucleus might be broken into an alpha particle, a proton and two neutrons. Certain elastic reactions other than the (n,n) reaction also may occur, usually with very high-energy neutrons.

NEUTRON INTERACTIONS WITH MATTER

As explained, any one of a number of possibilities of reactions can occur following bombardment of a material by a neutron, with some being more likely than others, depending on the specific situation. For example, when a ^{10}B nucleus is bombarded by a neutron, the compound nucleus (^{11}B) formed theoretically can disintegrate or decay in one of several ways (Figure 24-2-[illegible]).

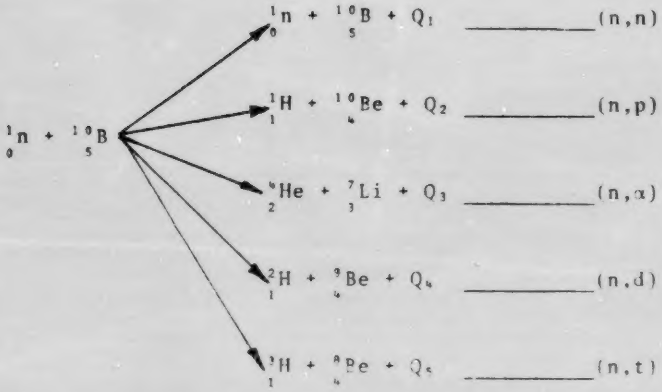

$^{1}_{0}n + ^{10}_{5}B$
$^{1}_{0}n + ^{10}_{5}B + Q_1$ (n,n)
$^{1}_{1}H + ^{10}_{4}Be + Q_2$ (n,p)
$^{4}_{2}He + ^{7}_{3}Li + Q_3$ (n,α)
$^{2}_{1}H + ^{9}_{4}Be + Q_4$ (n,d)
$^{3}_{1}H + ^{8}_{4}Be + Q_5$ (n,t)

Figure 24-2-9. Possible [$^{11}_{5}$B] disintegration methods.

Breakup (decay) of a compound nucleus is related to total energy of the system. Nuclei liberating larger values of Q have a greater likelihood of taking place, while those with negative values of Q must have at least that much energy supplied by a neutron for reaction to occur. Thus, if Q_2 and Q_4 are calculated, and $Q_2 = 0.738$ MeV while $Q_4 = -4.36$ MeV, a neutron must supply at least 4.36 MeV to Q_4 to make the (n,d) reaction occur. Otherwise, energy would not be conserved.

When Q is negative and energy must be supplied to make the reaction occur, the reaction is called endoergic. When Q is positive (i.e., energy is released), the reaction is termed exoergic.

The rule (the greater the Q, the more likely the reaction) is only generally true. Other than energy considerations, various compensated aspects such as potential barrier of the nucleus and thermodynamics of the system must to be evaluated. The problem is highly complex and difficult, but the Rule of Q is adequate for most approximations.

CROSS SECTIONS

Cross section is a measure of the probability of an interaction. A neutron cross section is expressed in units of area, usually cm^2. Because the effective area of a nucleus for a particular nuclear reaction is small, a new unit of area was created to eliminate the use of large negative exponents. This unit "barn," is equal to 10^{-24} cm^2 — approximately the cross-sectional area of a typical nucleus.

Many factors influence a cross section:

Type of reaction.
Energy of incident neutron.
Material undergoing reaction.

Cross section, commonly expressed in barns per atom or microscopic cross section, is symbolized by σ. There is a microscopic cross section for each particular type of reaction.

Total cross section, the sum of all other cross sections (e.g., scattering, capture, fission, etc.), can be expressed mathematically:

$$\sigma_{tot} = \sigma_{scattering} + \sigma_{capture} + \sigma_{fission} + \cdots$$

or

$$\sigma_t = \sigma_s + \sigma_c + \sigma_f. \qquad \text{Eq. 24-2-5}$$

Certain types of cross sections tend to predominate at specific energies, and also may show recognizable trends as a function of energy. The greater the neutron energy, the lower the capture cross section:

$$\sigma_c \propto \frac{1}{v} \qquad \text{Eq. 24-2-6}$$

where

v = velocity of neutron.

There is a greater cross section with neutron velocity; consider the slower neutrons as spending more time in the vicinity of the nucleus and having proportionately more time to interact. Because the kinetic energy is given by $E = \frac{1}{2}mv^2$ (and $v = \sqrt{\frac{2E}{m}}$), capture cross section is related to energy:

$$\sigma_c \propto \frac{1}{v}$$

$$\sigma_c \propto \frac{1}{\sqrt{\frac{2E}{m}}} = \frac{1}{\sqrt{\frac{2}{m}} \times \sqrt{E}} \qquad \text{Eq. 24-2-7}$$

But, $\frac{2}{m}$ is a constant, therefore, it does not affect proportionality. And cross section is a function of energy:

$$\sigma \propto \frac{1}{\sqrt{E}} \qquad \text{Eq. 24-2-8}$$

Neutron cross sections often are plotted as a function of neutron energy for a particular nuclide. The plot is generally a log-log plot; that is, the log of the cross section (σ) is plotted against the log of the neutron energy (E). This arrangement permits several magnitudes of energies and cross sections to be shown on a relatively small, smoothed, hypothetical plot (Figure 24-2-10). In general, the $\frac{1}{v}$ relationship seems to be followed. Irregularities appear at various points of the curve: 1) approximately 10^{-6}, 2) 3×10^{-5}, 3) 1 and 4) 5 MeV. (Neutrons at these specific energies have a much greater chance of interacting than do neighboring neutrons of slightly greater or lesser energies.) The sharp rises in cross section are known as resonances; the prime reactions that occur at these specific points are resonance reactions.

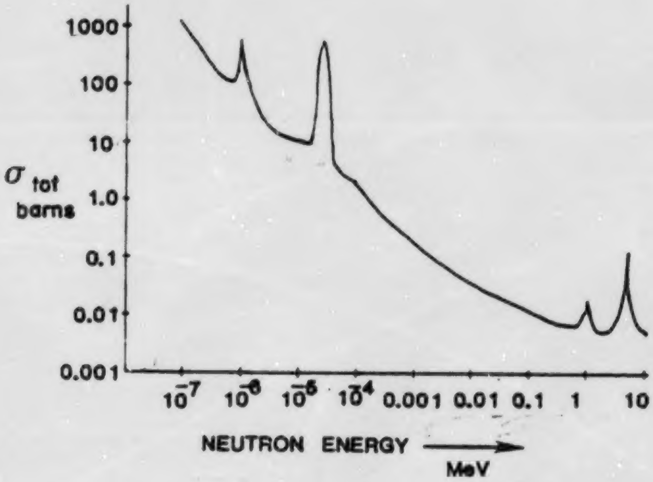


Figure 24-210.
Neutron cross section versus neutron energy.

Resonances are a function of the energy levels within the nucleus. Some elements, such as boron, have few resonances and follow very closely the $\frac{1}{v}$ relationship over a wide energy range. Others, such as cadmium and indium, have many resonances; their cross-section spectra are quite complex. For example, the capture cross section of cadmium shows a resonance for neutrons with energies about 0.176 eV. (At 0.176 eV, the cross section is 7200 barns. At 1 eV, the cross section is only 20 barns, and at 10 eV, about 4 barns.) Indium shows 3 resonance peaks between 1 and 10 eV, and 8 peaks between 10 and 100 eV (Figure 24-2-11). The largest of the cross-section peaks, at 1.4 eV, has a value of about 25,500 barns.

Indium and cadmium are good neutron absorbers, particularly for slow and thermal neutrons. Thus, they are sometimes used as thermal neutron shields, or as poisons to absorbe neutrons in a reactor. Boron is also a good neutron absorber, but it has few resonanaces.

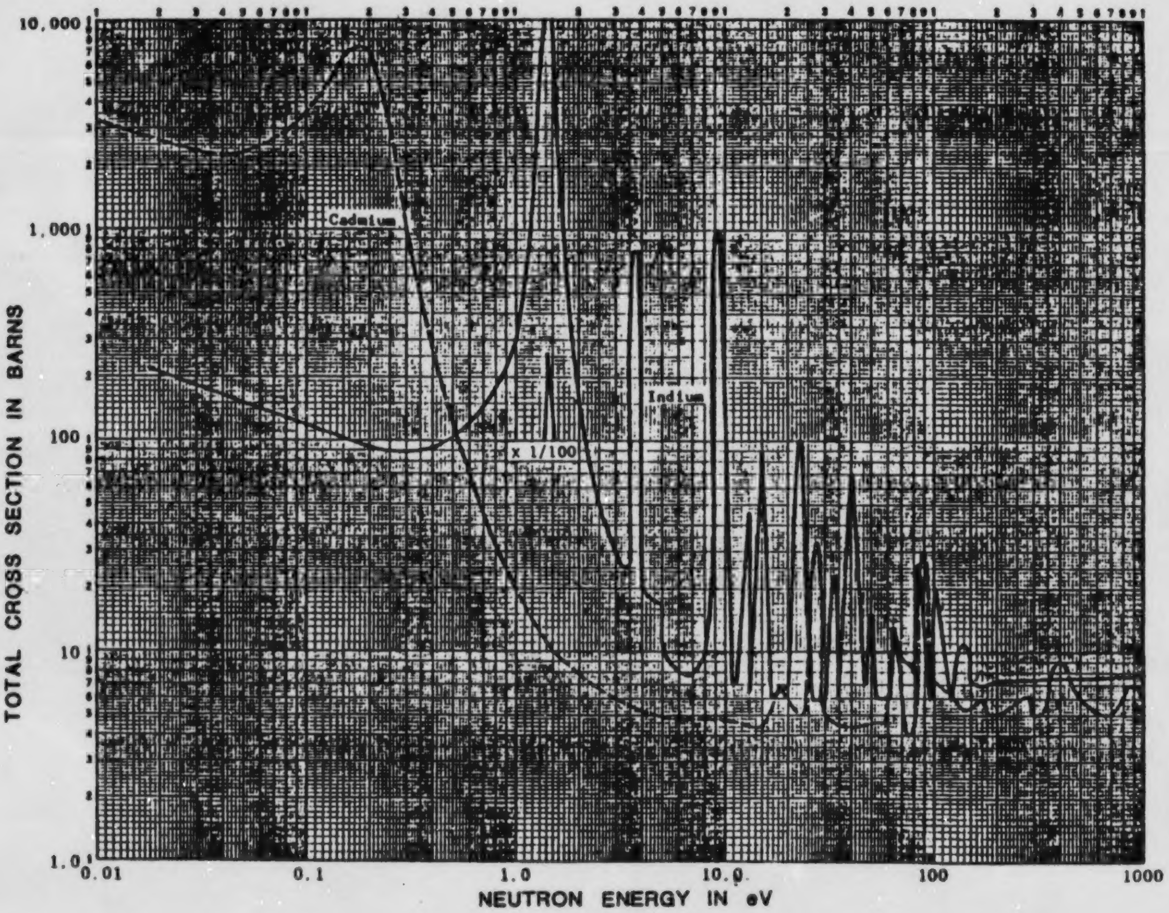


Figure 24-2-11. Typical neutron cross sections.

In general, capture cross sections predominate at low and intermediate energies, while particle reactions [(n,p), (n,α), etc.] tend to predominate at higher energies.

Neutron cross sections can be measured in various ways.

The most common way is to insert a thin foil of very pure material in a beam of mono-energetic (single-energy) neutrons, then measuring, after a known exposure time, radioactive products in the foil. This method is known as neutron activation. Total cross section also can be obtained by direct measurement of the neutron beam (Figure 24-2-12). This total cross section, called removal cross section,

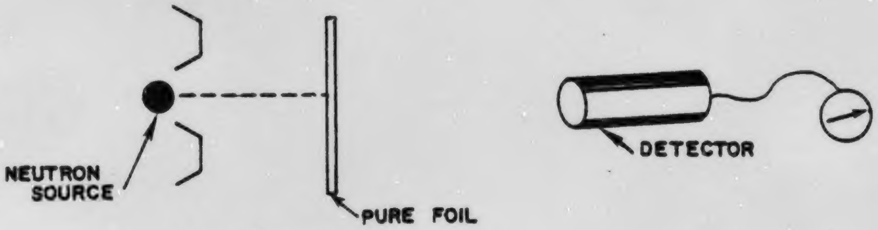


Figure 24-2-12. Neutron cross section measurement.

removes neutrons from the beam, either by scattering or by capture. Removal cross section, σ, can be calculated by the relationship:

$$I = I_o d^{-N\sigma x} \qquad \text{Eq. 24-2-9}$$

where

I = flux after passing through foil.
I_o = flux (at same point) with no foil.
N = number of atoms per cm^3 in foil.
x = thickness of foil.

Equation 24-2-9 has the same form as the equation for attenuation of gamma rays and is used in much the same manner. As an illustration, consider the following problem:

EXAMPLE

Given: A sheet of gold, 0.01 cm thick, inserted into a neutron beam, with a flux measurement of 640 n/cm^2-s; when foil is removed, flux-reading jumps to 880 n/cm^2-s. Also, the density (ρ) of gold is 19.32 g/cm^3 and its atomic weight is 197.

Find: Removal cross section N of gold.

Solution:

$N = \rho$ x Avogadro's number ÷ atomic weight

$$= \frac{19.32 \text{ g/cm}^3 \times 6.023 \times 10^{23}}{197 \text{ grams/gram atomic weight}} \times \frac{\text{atoms}}{\text{grams atomic weight}}$$

$$= \frac{1.932 \times 10 \times 6.023 \times 10^{23}}{.97 \times 10^2} \times \frac{\text{atoms}}{\text{cm}^3}$$

$$= 5.91 \times 10^{22} \text{ atoms/cm}^3.$$

Use Equation 12-2-9.

$$I = I_o e^{-N\sigma x}.$$

$$I = 640 \text{ n/cm}^2\text{-s}.$$

$$I_o = 880 \text{ n/cm}^2\text{-s}.$$

$$x = 0.01 \text{ cm}.$$

$$N = 5.91 \times 10^{22}\ \frac{\text{atoms}}{\text{cm}^3}.$$

$$640 = 380\ e^{-5.91 \times 10^{22}\ \frac{\text{atoms}}{\text{cm}^3} \times \sigma \times 0.01\ \text{cm}.}$$

Substitute and solve for σ:

$$\frac{640}{880} = e^{\ -5.91 \times 10^{22}\ \frac{\text{atoms}}{\text{cm}^3} \times \sigma \times 10^{-2}\ \text{cm}.}$$

Take ℓn of both sides:

$$\ell n \propto \left(\frac{640}{880}\right) = -5.91 \times 10^{20}\ \frac{\text{atom}}{\text{cm}^2} \times \sigma$$

$$\ell n\ .727 = -0.319$$

$$= \sigma = \frac{-0.319}{-5.91 \times 10^{20}\ \text{atom/cm}^2}$$

$$\sigma = 5.39 \times 10^{-22}\ \text{cm}^2$$

$$\sigma = 539\ \text{barns}\ (1\ \text{barn} = 10^{-24}\ \text{cm}^2).$$

The calculation of N, the number of atoms per cm^3, can become complex and difficult if compounds or mixtures rather than single pure elements are involved. It sometimes is more convenient and useful, therefore, to use a different type of cross section — the so-called macroscopic cross section — to eliminate the need for calculating N.

MACROSCOPIC CROSS SECTION

A macroscopic cross section, symbolized by Σ, is mathematically equal to the product of N and σ:

$$\Sigma = N\sigma. \qquad \text{Eq. 24-2-10}$$

Thus, the macroscopic cross section has units of reciprocal (or inverse) length and usually is expressed as cm^{-1} or "per centimeter." It may be thought of as the probability of a neutron interaction per unit length (or cm) of a material. In addition, there are individual macroscopic cross sections for caputre (Σ_c), scattering (Σ_s) and fission (Σ_f).

The cross section, σ (called microscopic cross section to distinguish it from macroscopic cross section, Σ) refers to one atom. It is dependent only upon the material and neutron energy. Macroscopic cross section, on the other hand, refers to the cross section of one cm^3 of atoms. It depends on atomic density, on the different materials present, and on neutron energy.

An important application of macroscopic cross section is in neutron shielding calculations. Materials such as paraffin and polyethylene contain many elements and, rather than attempt to determine multiple microscopic cross sections, a macroscopic cross section is employed in these calculations.

EXAMPLE

Given: A hypothetical substance (polyplastin) is inserted in a neutron beam and the flux measured as 166 n/cm^2-s; when removed from the beam, the flux was measured as 283 n/cm^2-s; polyplastin is 3.6 cm thick.

Find: Macroscopic cross section of polyplastin.

Solution:

$$\Sigma = N\sigma$$

and

$$I = I_o e^{-N\sigma x}.$$

Substituting from Equation 12-2-10:

$$I = I_o e^{-\Sigma x}$$

$$166 = 283E^{-\Sigma \text{ x } 3.6 \text{ cm}}$$

Solving for Σ:

$$\frac{166}{283} = e^{-\Sigma \text{ x } 3.6}$$

$$\ell n \left(\frac{166}{283}\right) = -3.6 \text{ x } \Sigma$$

$$\ell n\ 0.59 = -3.6\Sigma$$

$$-0.534 = -3.6\Sigma$$

$$\Sigma = \frac{-0.534}{-3.6} = 0.15 \text{ cm}^{-1}$$

e-FOLDING THICKNESS

A shielding term sometimes used, e-folding thickness, is the amount of shielding required to reduce neutron flux by a factor of e (the base of the natural logarithms, numerically equal to 2.718282). If a neutron beam is reduced by a factor of e, it is reduced to $\frac{1}{e}$ = 0.37, or 37% of its original intensity.

Of perhaps more significance is the calculation of the cross sections for individual reactions. These are important for many reasons, one being that knowledge of these cross sections permits calculation of induced radioactivity (and hence dose rates).

Earlier in this section of the module, the cross section for a particular element in a neutron beam was calculated. What was calculated was σ_{tot}, the total microscopic cross section, which is the sum of the cross sections for all the individual neutron reactions:

$$\sigma_{tot} = \sigma_{scattering} + \sigma_{n,\alpha} + \sigma_{fission} + \sigma_{n,p} + \ldots$$

There are several ways to determine individual cross sections, but the easiest is by measurement of one (or more) of the products of the reaction. Since these are usually radioactive, this measurement often can be made easily.

Suppose, for example that the cross section for a (n,α) reaction in aluminum is desired:

$$^{27}_{13}\text{Al} + {}^{1}_{0}\text{n} \rightarrow {}^{24}_{11}\text{Na} + {}^{4}_{2}\text{He} + \text{Q}. \qquad \text{Eq. 24-2-11}$$

The product ^{24}Na is radioactive (having a 15-hour half-life and emitting two gammas per disintegration, one with an energy of 1369 keV and the other 2.754 keV). If only gamma rays are sought, then only the amount of ^{24}Na is measured, and not other radioactive species, such as ^{28}Al or ^{28}Mg, or the products of other possible reactions.

The equation used to calculate $\sigma_{(n,\alpha)}$ is derived from the equation used to calculate total removal cross section. But it has somewhat different form because radioactivity induced by neutrons is being evaluated (instead of a direct measurement of flux change).

To find out the (n,α) in aluminum, a small amount (say 1 gram) is placed in a neutron flux of 10^6 n/cm^2-s and left there for 100 minutes. It then is removed from the neutron flux and carried to the laboratory, where two hours later the radioactivity is measured from ^{24}Na as 410 disintegrations per second. Then $\sigma_{(n,\alpha)}$ is calculated as follows:

$$A = n\sigma\phi\ (1-e^{-\lambda\tau})e^{-\lambda t} \qquad \text{Eq. 24-2-12}$$

where

A = activity of sample, disintegration/second.
n = number of atoms.
σ = microscopic cross section, cm^2.
ϕ = neutron flux, neutrons/cm^2-s.
λ = decay constant.
τ = duration time.
t = time (removal from exposure until measurement).

Figure 24-2-13 gives a graphical presentation of build-up and decay. Note that, in the above calculation, all units of time were kept in seconds, and the equation is dimensionally correct:

$$\frac{\text{atoms (disintegrating)}}{\text{SECOND}}$$

$$= \text{atoms} \times \frac{\text{cm}^2}{\text{n}} \times \frac{\text{n}}{\text{cm}^2\text{-s}}\ (1-e^{-s^{-1}} \times s)\ (e^{-s^{-1}} \times s)$$

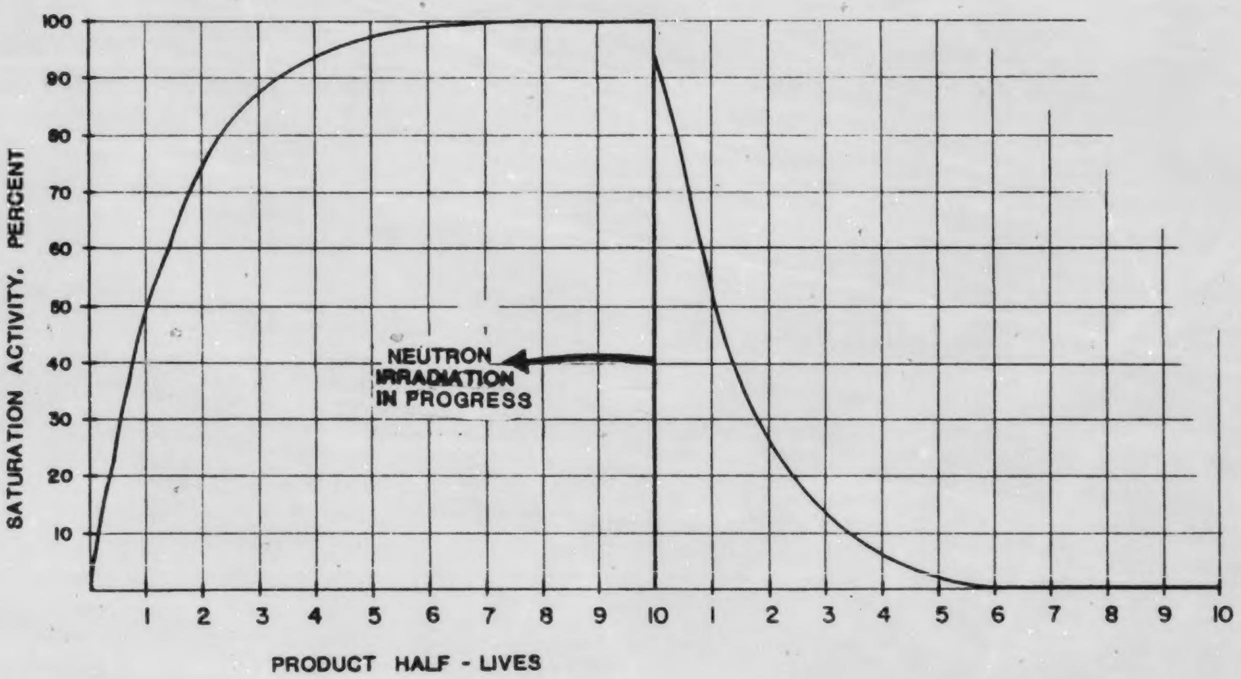


Figure 24-2-13. Relationship of activity buildup and decay of radioactive product as a result of neutron activation.

PRACTICAL APPLICATION OF CROSS SECTIONS

Generally, cross sections are known, and the above equations given are used frequently for neutron shielding calculations; values for σ_{tot} or Σ are available from handbooks, most notably, Neutron Cross Sections, BNL-325, the so-called "barn book," which is a compilation of graphs of σ versus neutron energy for many elements and materials.

A most important application of cross sections is their use in calculating induced radioactivity from neutron bombardment, using Equation 24-2-12:

$$A = n\sigma\phi\ (1-e^{-\lambda\tau})\ e^{-\lambda t}.$$

EXAMPLE

Given: A source of ^{60}Co to be produced in a high flux research reactor; thermal flux is 10^{14} n/cm-s; ^{59}Co target mass is 10 g.

Find: How long one would have to expose the sample of ^{59}Co to obtain 1 Ci.

Solution: From a handbook or Chart of the Nuclides, find thermal neutron capture cross section of ^{59}Co. (For convenience, use 37.4 barns.)

1 Ci = 3.7×10^{10} atoms disintegrating per second, which is the value of A (in Equation 12-2-12). Number of atoms, n, is calculated as follows:

$$n = \frac{10 \text{ grams} \times 6.023 \times 10^{23} \ \frac{\text{atom}}{\text{g-at wt}}}{59 \text{ grams/g-at wt}}$$

$$= 1.02 \times 10^{23} \text{ atoms.}$$

The decay constant, λ, is calculated from the half-life of ^{60}Co:

$$\lambda = \frac{0.693}{5.35 \text{ yr} \times 3.15 \times 10^{7} \text{ s/yr}} = 4.1 \times 10^{-9} \text{ s}^{-1}$$

If distance is zero, to produce 1 Ci or 3.7×10^{10} dis/s at instant of removal from reactor, time after removal from reactor is zero; hence, $e^{-\lambda t} = e^{o} = 1$. Now, the only unknown is τ (irradiation time); values can be substituted in Equation 24-2-12 to solve for τ:

$$A = n\sigma\phi(1-e^{-\lambda\tau})\ e^{-\lambda t}$$

$$3.7 \times 10^{10} \text{ dps} = 1.02 \times 10^{23} \text{ atoms} \times 37.4 \times 10^{-24} \text{ cm}^2/\text{atoms}$$

$$\times\ 10^{14} \text{ n/cm}^2\text{s}$$

$$\times\ (1-e^{-4.1 \times 10^{-9}/\text{s}\ \tau})(e^{-4.1 \times 10^{-9} \text{ s} \times 0 \text{ s}}).$$

Therefore,

$$3.7 \times 10^{20} \text{ dps} = 3.81 \times 10^{14} \text{ atoms/s } (1-e^{-4.1 \times 10^{-9}\tau}) \times 1$$

$$9.71 \times 10^{-5} = 1-e^{-4.1 \times 10^{-9}\tau}$$

$$9.71 \times 10^{-5} -1 = -e^{-4.1 \times 10^{-9}\tau}$$

$$\ell n\ 0.999903 = -4.1 \times 10^{-9}\tau$$

$$\frac{-9.7 \times 10^{-5}}{-4.1 \times 10^{-9}} = \tau$$

or

$\tau = 2.37 \times 10^{4}$ seconds
$= 3.95 \times 10^{2}$ minutes
$= 6.58$ hours.

An important concept related to cross section is that of albedo. Neutron albedo refers to reflected neutrons and is, therefore, related to the scattering cross section. Neutron reflectors generally are low-Z materials that are also good moderators. Such materials have high scatter cross sections and can reflect (or bounce) neutrons back into a reactor. Neutron albedo often is expressed as a fraction of the total neutron flux.

A final concept related to cross section is cadmium ratio. Cadmium has a very high capture cross section for thermal neutrons, in the neighborhood of several thousand barns. A crude but effective indication of how well a neutron flux or beam is thermalized can be obtained by making a measurement with a bare detector or foil and comparing it with a detector or foil that has been wrapped in cadmium. Usually, two gold or indium foils are used, one bare and one wrapped in a sufficient thickness of cadmium to remove 99+% of the thermal neutrons. A cadmium thickness of less than 0.001 inch is adequate to accomplish this. Hence, a bare foil and one wrapped in some cadmium (both foils with the same mass) can be placed in a neutron flux, exposed for the same amount of time, and radioactivity in each foil determined. The ratio of activation (i.e., radioactivity) in the bare foil to that in the cadmium-wrapped foil is the cadmium ratio:

$$\text{CD ratio} = \frac{\text{activity in bare foil}}{\text{activity in Cd-wrapped foil}}$$

Eq. 24-2-13

REACTION RATE

Although there are many potential neutron reactions, the most important from a reactor standpoint are those collisions that reduce the energy of fast neutrons and thereby produce thermal neutrons. These reactions are primarily elastic scattering. When fast neutrons, having energies on the order of a few MeV, are introduced into a medium, such as water, which consists largely of atoms with nuclei about the same size as the neutron, a sequence of collisions between the neutrons and the nuclei occurs. With each collision, the neutrons lose energy and change their direction of travel (Figure 24-2-14). The x's refer to collisions with other (hydrogen) nuclei.

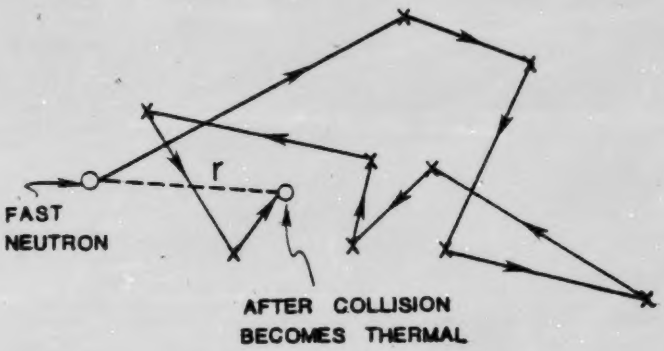


Figure 24-2-14. Typical path during slowing down of fast neutron. (Distance from point of origin to point of thermalization is shown by dash line, r.)

The average elastic scattering collision is a function of two characteristics that depend only on the nucleus involved in the scattering:

The mean of the cosines of the angles of scattering.

The mean change in the natural logarithm of the energy of the neutron.

Each of these characteristics can be expressed mathematically, and the mathematical solutions demonstrate that light nuclei, such as hydrogen, tend to scatter neutrons in the forward direction and to provide the greatest energy loss per collision. Thus, a neutron passes through a series of collision, traveling different distances between collisions and ultimately, after many collisions, stopping at some distance r from its point of origin as a thermal neutron.

This distance r, which is important in reactor work, is related to a concept known as neutron age. This age, given in units of area (actually, square centimeters), is related to the average distance (hence, time) between point of origin and point of thermalization. Obviously, the smaller the age, the better the material is at slowing down neutrons. Materials that slow down neutrons are called moderators; Table 24-2-1 shows the "age" for three commonly-used moderators.

Neutron reaction rate R, a function of three variables, can be expressed mathematically:

$$R = \phi N \sigma. \qquad \text{Eq. 24-2-14}$$

where

R = reaction rate, reactions (or neutrons)/cm^3-s.

ϕ = neutron flux, neutrons/cm^2-s.

N - number of target atoms per cm^3.

σ = cross section per target atom, cm^2.

TABLE 24-2-1
AGE OF MODERATORS

Moderator	"Age" to Thermalization
Light water	26 cm^2
Heavy water	125 cm^2
Graphite	365 cm^2

Obviously, the greater the neutron flux, the greater the reaction rate. Similarly, the greater the cross section or number of target atoms, the greater the reaction rate.

Note that reaction rate has units of n/cm^3-s (neutrons reacting per cubic centimeter per second). Also, reaction rate can be expressed in terms of Σ, the macroscopic cross section, because, as was shown above,

$$R = \phi\Sigma \qquad \text{Eq. 24-2-15}$$

To illustrate the reaction rate, the microscopic fission cross section for ^{235}U is 681 b (for thermal neutrons), and the corresponding macroscopic fission cross section, $\Sigma_a = 32.7\ cm^{-1}$. In a small research reactor, where the thermal flux level is 4×10^{13} neutrons/cm^2-s in the center of the core, the reaction rate may be shown as follows:

$$R = \phi\Sigma = 4 \times 10^{13}\ \text{neutrons}/cm^2\text{-s} \times 32.7\ cm^{-1}$$
$$= 1.31 \times 10^{15}\ n/cm^3\text{-s}.$$

This rate is the same as that at which ^{235}U atoms are consumed, since they are transformed (or fissioned) by the reaction. Thus, reaction rate can be used to determine fuel consumption in a reactor.

EXERCISES

1. Write equations for the following reactions:
   - Neutron capture by ^{59}Co.
   - (n,α) or ^{14}N.
   - $(n,2n)$ or ^{13}C.
2. Calculate Q for an (n,α) or ^{14}N reaction. Would this reaction take place with 2-MeV neutrons (according to the Rule of Q)?
3. Use Figure 24-2-11; determine resonance cross section for 38-eV neutrons in indium.
4. An accelerator produces a neutron beam with a flux of 3.6×10^{14} n/cm^2-s; when a sheet of indium is inserted in the beam, flux drops to 2.1×10^{14} n/cm^2-s; if indium is 0.6 mm thick, calculate macroscopic cross section for removal. What is its microscopic cross section?
5. Thermal neutron capture cross section for gold is 99 barns; a one-gram foil of gold is inserted into a reactor and exposed to a thermal neutron flux of 1.6×10^{12} n/cm^{-2}-s for three hours; it is removed and measured for radioactivity one week later. How much radioactivity could be expected to be found?
6. A neutron shielding material has a macroscopic cross section for fission neutrons of 8.7 cm^{-1}. How many

cm would be required to reduce a fission neutron flux of 6400 n/cm^2-s to 7.5 n/cm^2-s?

MATERIALS

1 neutron source.
Cadmium sheets and foils.
Gold foils.
1 neutron survey meter.
1 scaler and GM tube (counting equipment).
Paraffin or polyethylene sheets.
Semi-log paper.

PROCEDURES

NEUTRON ATTENUATION

Measure output from a neutron source with a survey meter. Plot change in measured output on semi-log paper, as several cadmium sheets of known thickness are successively inserted, one at a time, between source and survey meter. Calculate macroscopic cross section (for this particular source) for cadmium. Record all data on Data Table 24-2-1.

NEUTRON ACTIVATION

Place bare gold foil and a gold foil wrapped in cadmium at equal distances from neutron sources. Expose for an appropriate period (specified by instructor). With a laboratory scaler and GM tube, determine counting rates for bare and covered foils. Calculate Cd ratio. Also, calculate thermal capture cross section for gold (σ_c). Assume that all activation is caused by thermal neutrons. Record all data on Data Table 24-2-2.

Repeat this experiment with a sheet of polyethylene (or paraffin several inches thick) between source and foils. (Alternatively, foils may be imbedded in hydrogenous material.) Determine Cd ratio and explain why it varies from previous one.

DATA TABLES

DATA TABLE 24-2-1

Type neutron source ______________________________

Model number of survey meter ________________________

Distance from source to survey meter _________________

Thickness of Cadmium	Survey Meter Reading
0	

DATA TABLE 24-2-1 (continued)

Directions on graph: Plot survey meter reading on vertical (semi-log) axis, and thickness on horizontal (linear) axis. Draw a straight line through data, extending line back to X = 0. From straight line, not actual data, pick off the following information:

I_o = ________________, at X = 0.

I = ________________, at X = _____________.

(Pick some arbitrary I and a corresponding X.)

Solve for Σ:

How do experimental values compare with accepted values (given by instructor)?

70

DATA TABLE 24-2-2

Type neutron source ______________________________

Model number of scaler and tube ______________________

Mass of bare gold foil ___________________________

Mass of covered gold foil _________________________

Activity in bare foil (net) _________________________

Activity in Cd covered foil (net) ___________________

Note: If foil masses are different, a correction must be made because activity is proportional to mass:

$$\text{Cd ratio} = \frac{\text{(activity in bare foil}}{\text{(activity in Cd-wrapped foil)}} \; \frac{\text{(mass of Cd-wrapped foil)}}{\text{(mass of bare foil)}}$$

$$= \frac{(\qquad\qquad\qquad\qquad)}{(\qquad\qquad\qquad\qquad)} \; \frac{(\qquad\qquad\qquad\qquad)}{(\qquad\qquad\qquad\qquad)}$$

$$= \underline{\qquad\qquad\qquad\qquad\qquad}.$$

Calculate thermal capture cross section:

How do experimental values compare with accepted values (given by instructor)?

DATA TABLE 24-2-3

Type neutron source ______________________________

Model number of scaler and tube ____________________

Mass of bare gold foil ___________________________

Mass of covered gold foil _________________________

Activity in bare foil (net) ________________________

Activity in Cd-covered foil (net) ___________________

Note: If foil masses are different, a correction must be made because activity is proportional to mass.

$$\text{Cd ratio} = \frac{\text{(activity in bare foil)}}{\text{(activity in Cd-wrapped foil)}} \; \frac{\text{(mass of Cd-wrapped foil)}}{\text{(mass of bare foil)}}$$

$$= \frac{(\qquad\qquad\qquad)}{(\qquad\qquad\qquad)} \; \frac{(\qquad\qquad\qquad)}{(\qquad\qquad\qquad)}$$

$$= \underline{\qquad\qquad\qquad\qquad}.$$

Calculate thermal capture cross section:

Explain why Cd ratio and thermal neutron cross section vary from previous case.

How do experimental values compare with accepted values (given by instructor)?

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TEST

1. List three different types of reactions that produce neutrons:

2. Write typical reaction equations for radioactive capture and charged particle emission:

3. Listed below are four neutron scattering reactions. Indicate process that takes place during reaction:

   Head-on collision with a small nucleus.

   Glancing collision with a small nucleus.

Head-on collision with a large nucleus.

Glancing collision with a large nucleus.

4. Briefly describe concept of cross section:

5. Why is cross section not equal to cross-sectional area of a nucleus?

6. Use "Chart of the Nuclides" and Radiological Health Handbook as references; define:

   Scattering.

Absorption.

Fission cross sections.

7. Briefly explain differences between microscopic and macroscopic cross sections:

8. Use radionuclide and density; calculate macroscopic cross section for ^{12}C (find necessary values in "Chart of the Nuclides"):

9. Draw a typical absorption-versus-energy curve:

10. Briefly explain significiance of major parts of curve drawn above:

11. Define the following terms:
    a. Nuclear reaction.

    b. Elastic scattering.

    c. Inelastic scattering.

    d. Capture.

    e. Compound nucleus.

    f. Fission.

    g. Cross section.

    h. Microscopic cross section.

i. $\frac{1}{v}$.

j. Resonance.

k. Macroscopic cross section.

l. Albedo.

m. Cadmium ratio.

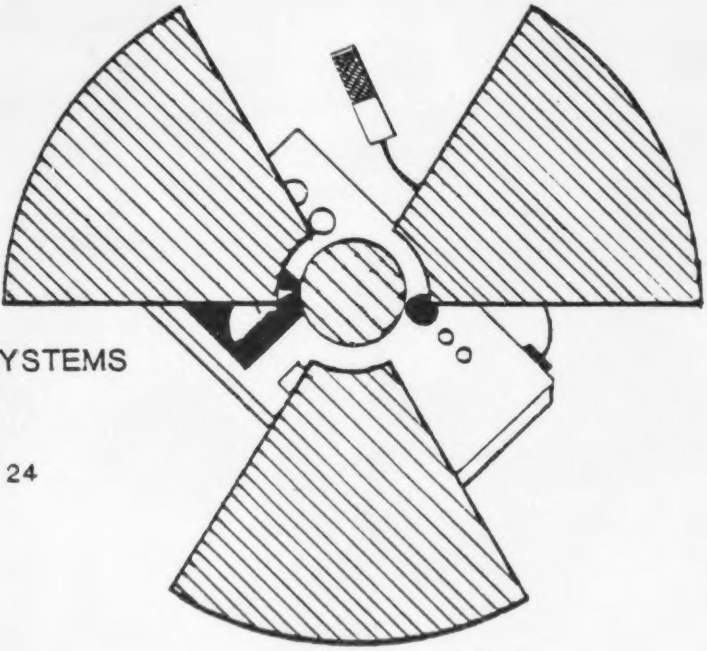
n. Neutron age.

o. Neutron reaction rate.

NUCLEAR TECHNOLOGY

MODULE 24 - 3

FISSION RATE AND REACTOR POWER



NUCLEAR SYSTEMS
AND SAFETY

COURSE 24

CENTER FOR OCCUPATIONAL RESEARCH AND DEVELOPMENT

COURSE MODULES

COURSE 24 NUCLEAR SYSTEMS AND SAFETY

MODULE 24-1	Neutron Physics
MODULE 24-2	Cross Sections and Reactions
MODULE 24-3	Fission Rate and Reactor Power
MODULE 24-4	Nuclear Criticality (0530 ERDA Manual)
MODULE 24-5	Nuclear Material Inventory Management Techniques
MODULE 24-6	Nuclear Plant Safeguards

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INTRODUCTION

When certain heavy nuclei capture neutrons, the nuclei split into smaller nuclei. This reaction is called "fission." In addition to the two main fission products (smaller nuclei), two other products result from the fission process: (1) energy is released and (2) usually, two or three neutrons are released. These neutrons then may interact with other nuclei to cause additional fission, and a process (known as a chain reaction) ensues.

This module is a study of the energy (or heat) and power produced from such nuclear fission.

MODULE PREREQUISITES

The student should have completed Course 1, "Radiation Physics"; Course 2, "Radiation Protection I"; Course 11, "Radiation Detection and Measurement"; Course 17, "Radiation Protection II"; Course 23, "Nuclear Chemical Processes"; and the previous two modules of this course.

OBJECTIVES

Upon completion of this module, the student should be able to:

1. In one or two sentences, describe the important contributions of each of these men to the discovery of the fission process: (1) Fermi, (2) Hahn, and (3) Strassman.

2. State why it is easier to cause fission with neutrons than with other nuclear projectiles.
3. Differentiate between fissionable, fissile and fertile materials.
4. List the three most common fissionable nuclides.
5. Write the nuclear equations for converting ^{238}U into ^{239}Pu and ^{232}Th into ^{233}U.
6. Draw a curve for fission cross section versus neutron energy for ^{235}U.
7. Explain why most fission fragments are β^- emitters.
8. Draw and explain the significance of the fission yield curve for ^{235}U.
9. Calculate energy released in fission, given identity and masses of reactants and products.
10. Explain why the energy of a neutrino is not usable in a power reactor.
11. Discuss distribution of fission energy between the three major fission products.
12. Verify that 3.1×10^{10} fissions per second will produce 1 watt.
13. Given that 280 ^{235}U atoms spontaneously fission per second per ton of material, calculate how many tons are required to produce one watt of power from spontaneous fission.
14. Find the power at any time after shutdown, given a curve or equation relating the power and time after shutdown from full operating power.
15. Define:
    - Transuranium elements.
    - Fissile.
    - Fertile materials.

Fission yield.
Unsymmetrical yield.
Decay heating.
Spontaneous fission.

16. Calculate mass and volume of all fission products generated after a 1000-MW reactor has been operational for one year.

SUBJECT MATTER

DISCOVERY OF FISSION

In the mid-1930's, Enrico Fermi, working in Italy, exposed uranium ($Z = 92$) to low-energy neutrons (thermal or slow neutrons) with the intention of producing transuranium elements ($Z > 92$), which would be created by the (n,γ) reaction on the uranium followed by subsequent β^- decay. There were many different observed products, and this fact tended to confuse Fermi and others working on this approach. Then, in 1938, Otto Hahn and F. Strassman, German scientists, convincingly showed by chemical separation that most of the products were not transuranic elements, but rather were very much lighter than uranium with atomic numbers around 35 to 65. Clearly, for this to occur, the heavier uranium atom must have split (i.e., undergone fission).

Shortly thereafter, it was also observed that several neutrons were emitted as a part of the overall fission process. If the neutrons could be used to bombard more uranium, then more fission would occur and a chain reaction would ensue (Figure 24-3-1).

By 1942, Fermi had emigrated to the United States. On December 2, 1942, he and his coworkers at the University of Chicago demonstrated the first self-sustaining chain nuclear reaction under the stands at Stagg Field. The fuel was 40 tons of natural uranium mixed with blocks of graphite. This configuration resembled a pile of bricks and was given the name of a nuclear "pile," rather than nuclear "reactor," which is used today.

Shortly after the discovery of fission it was realized that, when uranium fissions, a considerable amount of energy

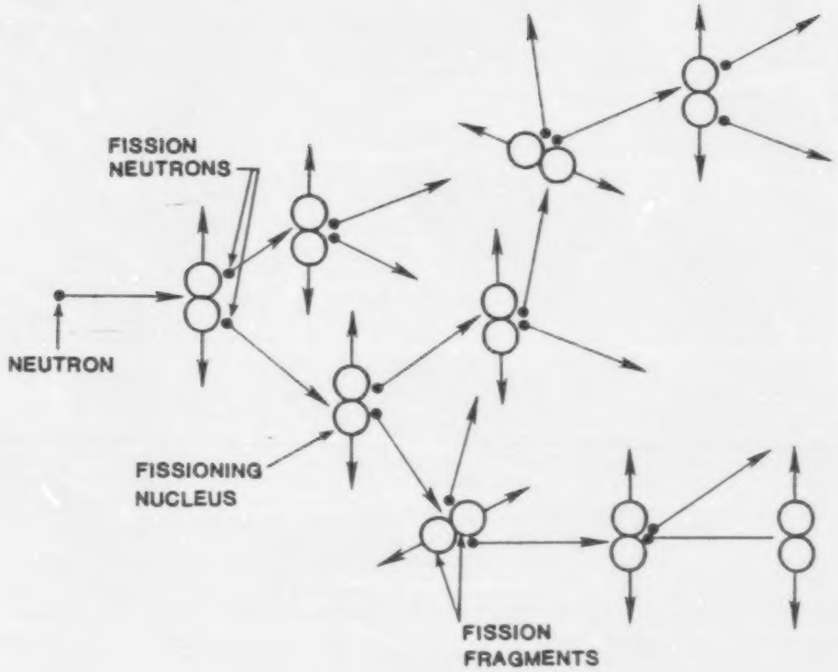


Figure 24-3-1. Fission chain reaction.

is released. The military significance of this energy was quickly realized by both the U.S. and German governments. The U.S. reaction was a successful crash program, called the Manhattan Project, to build a nuclear weapon.

So far, the major elements of the fission process have been outlined; that is, the two major inputs to the process have been identified (Figure 24-3-2): 1) a slow neutron and 2) a suitable heavy nucleus.

The three major products are 1) fission fragments, 2) neutrons and 3) energy. Let us examine each of these two processes and three products in much more detail.

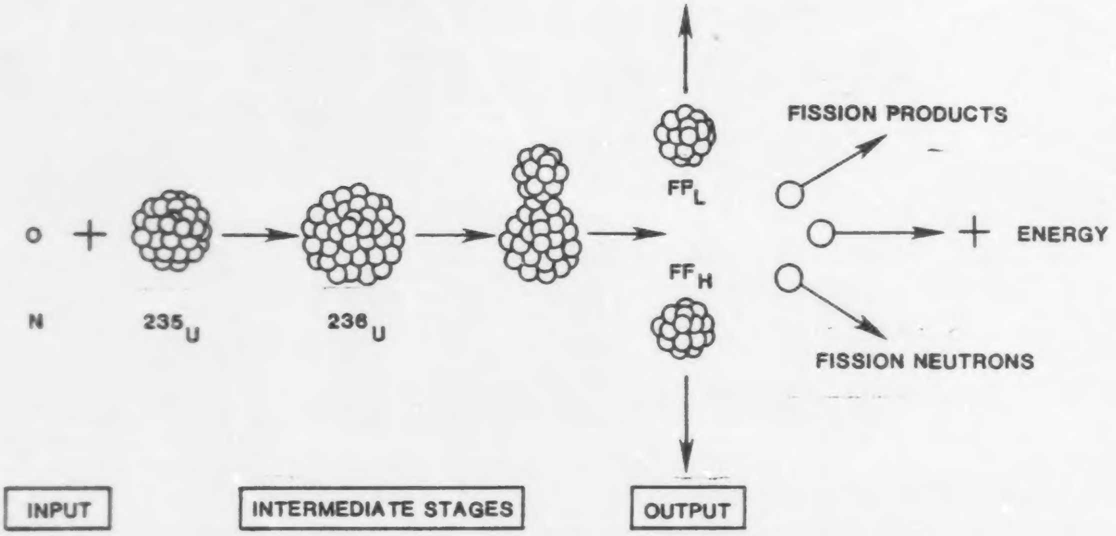

FISSION PRODUCTS
FP_L
ENERGY
FF_H
FISSION NEUTRONS
N
235U
236U
INPUT
INTERMEDIATE STAGES
OUTPUT

Figure 24-3-2. Four stages in fission of ^{235}U. (FF = fission fragments; H and L = heavy and light.)

FUEL

Only three nuclides exhibit the needed properties (of being both heavy and having an odd mass number) to be able to fission following the capture of a thermal neutron; three are ^{235}U, ^{233}U and ^{239}Pu. These nuclides, because they exhibit these special properties, are classified as fissile. Of the three, only ^{235}U is found in nature. The other two must be made from other nuclides. These other nuclides are classified as fertile materials (e.g., ^{238}U, ^{232}Th) because they exist in nature and fissionable materials can be made from them. Equations 24-3-1 and 24-3-2 give the pathways for the two transformations:

$$^{238}\text{U}\,(n,\gamma)\;{}^{239}\text{U} \xrightarrow[\beta^-]{} {}^{239}\text{Np} \xrightarrow[\beta^-]{} {}^{239}\text{Pu}$$

Equation 24-3-1
(Equation = Eq.)

$$^{232}\text{Th}\,(n,\gamma)\;{}^{233}\text{Th} \xrightarrow[\beta^-]{} {}^{233}\text{Pa} \xrightarrow[\beta^-]{} {}^{233}\text{U}$$

Eq. 24-3-2

An important point to be realized from the above discussion is that the only naturally-occurring fissile fuel is ^{235}U. It occurs in nature with an isotopic abundance of ~ 0.72%. The nonfissile, but fertile ^{238}U has an isotopic abundance of ~99.27%. A large fraction of the effort of the Manhattan Project was to separate and concentrate the ^{235}U from the ^{238}U.

All of this discussion is not meant to imply that ^{238}U or ^{232}Th will not fission. They will. But the necessary neutron energy must be of the order of 1 MeV. Therefore, these nuclides are classified as "fissionable." (They are not fissile because the energy spectrum of their fission neutrons is such that not all have sufficient energy to allow further fission and sustain a chain reaction.)

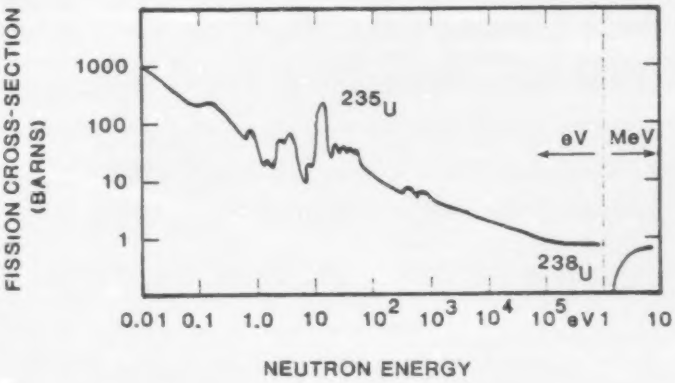
To summarize: A fissile nuclide undergoes fission, releasing neutrons, which can cause more fission. A fissionable nuclide can undergo fission, but will not necessarily sustain a chain reaction. A fertile nuclide, by definition, can be transformed into a fissile nuclide.

The interaction of a neutron with a uranium atom can occur at any neutron energy. The neutron is unlike any charged particle, such as a proton or alpha, which must have enough energy to overcome the coulombic barrier created between the charge of the heavy nucleus (92 protons in the case of uranium) and the charged particle. As will be shown in the next section, it is also highly desirable to have the incident neutrons at as low an energy as possible.

THERMAL NEUTRONS

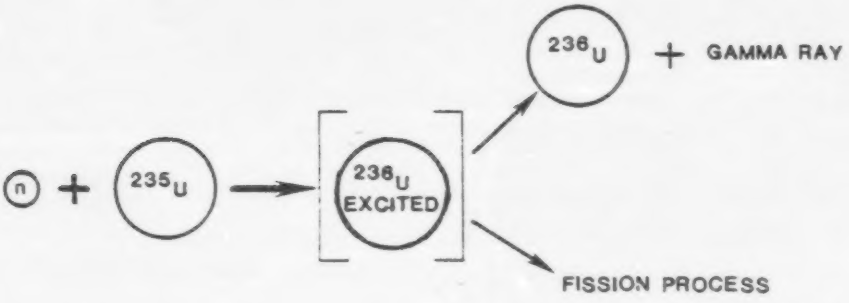
The fission cross section, σ_f in barns, as a function of the neutron energy in MeV (Figure 24-3-3) is an excitation function. As can be seen, the lower the energy, the more likely it is that fission will occur. As is shown, a reduction in energy by a factor of 10 increases the fission cross section by about 3.3 times. The irregular portion of the excitation function around 10 eV is attributed

Figure 24-3-3. Fission excitation function of ^{235}U.



to a phenomenon known as resonance absorption. Within this energy region, the neutron can easily fill excited energy levels in the ^{235}U that are not populated at lower neutron energies. Thus, the reaction cross section quickly rises and falls within the population of neutrons into these specific excited energy states.

Once the fissile nuclide has captured a neutron, it can de-excite by either gamma-ray emission or fission. Of the two pathways shown (Figure 24-3-4), only the fission route will be discussed here.



(as shown in Fig. 24-3-2)

Figure 24-3-4. Showing the competitive paths following capture of a neutron by ^{235}U.

FISSION FRAGMENTS

For the reaction shown earlier (Figure 24-3-2), fission fragments are spread between mass numbers 70 and 160, with a spread in atomic numbers between 35 and 65. While this spread occurs when many fission events are investigated, any one individual event must have products with a mass, including the two or three fission neutrons, of 236 and whose atomic number sum is 92. Thermal-energy-induced fission usually creates unequal fission fragment masses. The mass ratio of the heavy-to-light fragment is is ~ 3 to 2. In addition, neutron-to-proton ratio (N/Z) of the fission fragments reflects the (N/Z) ratio of the fissile parent. As a result, most fission products from ^{235}U have (N/Z) ratios around (235-92)/92 = 143/92 = 1.55.

Given these constraints, a typical, but by no means unique, set of fission fragments from ^{235}U would be ^{97}Sr and ^{137}Xe. The entire nuclear equation thus would be:

$$ {}^{1}_{0}n + {}^{235}_{92}U \rightarrow [{}^{236}_{92}U] \rightarrow {}^{97}_{38}Sr + {}^{137}_{54}Xe + 2n. $$

Eq. 24-3-3

These fission fragments are generally radioactive and decay through specific chains to stable isotopes.

The nuclear stability curve (Figure 24-3-5) shows that the common fragments from the fission of ^{235}U ($N/Z \approx 1.55$) rest above the stability line in the region of β^- decay. By this decay, N/Z ratio of the fission fragments is decreased to a point along the line of stability where further decay will not occur. The following example illustrates the process leading to stability and the relationship with the N/Z ratio.

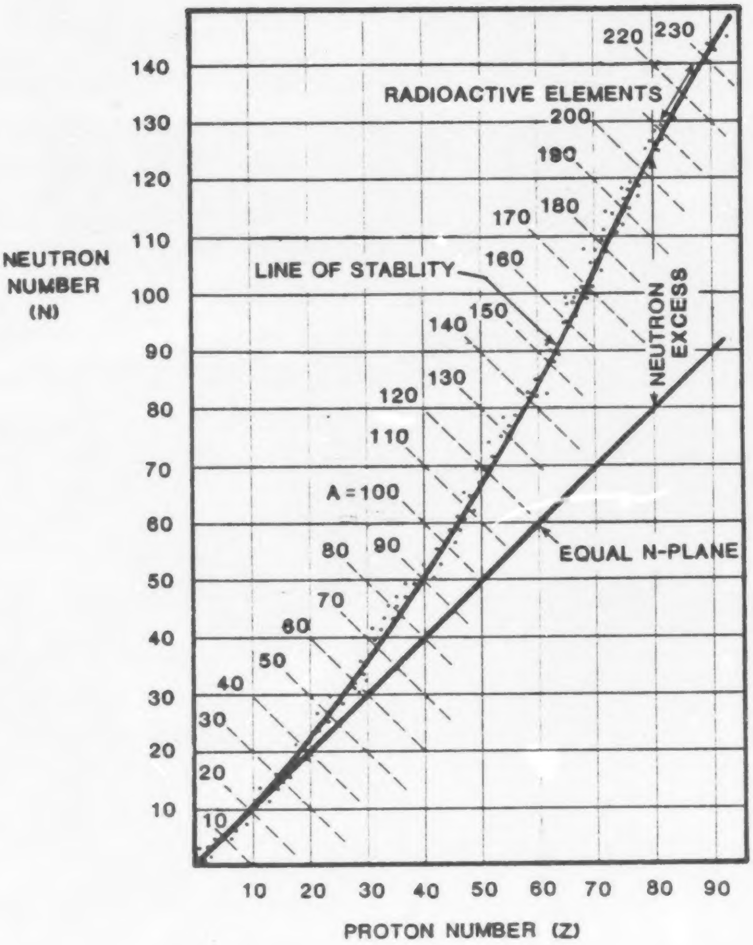


Figure 24-3-5
Nuclear stability
curve.

EXAMPLE

$^{137}_{52}\mathrm{Te} \xrightarrow[\beta]{} {}^{137}_{54}\mathrm{I} \xrightarrow[\beta^-]{} {}^{137}_{55}\mathrm{Xe} \xrightarrow[\beta^-]{} {}^{137}_{55}\mathrm{Cs} \xrightarrow[\beta^-]{} {}^{137}_{56}\mathrm{Ba}$ stable

N/Z ratio =

1.63 1.58 1.53 1.49 1.44

It is important to realize that there are two different numbers associated with the fission yield of each fragment nuclide. The independent yield is that resulting directly from fission; in other words, that which is formed at time of fission. Cumulative yield, on the other hand, of a fission fragment is the sum of the independent yield plus the yield resulting from any β^- decay, and is time dependent. In the example, 21% of the ^{137}Xe yield from ^{235}U fission is independent. The remainder (79%) comes via the β^- decay of ^{137}I.

A common way of presenting the yields of all the products is to plot the cumulative fission yield for the entire mass chain as a function of the mass number. If, in the above example, the cumulative fission yield for ^{137}Xe would be plotted at 137, a fission yield curve for thermal neutrons interacting with ^{235}U may be developed (Figure 24-3-6).

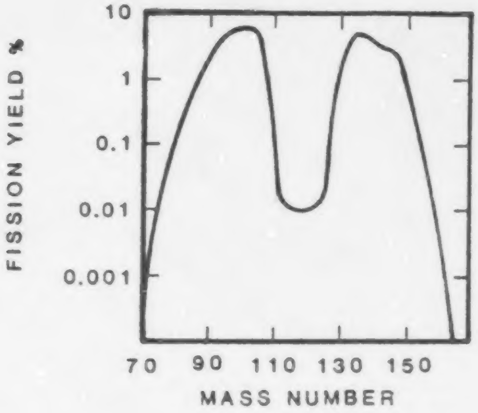


Figure 24-3-6.
Fission yield curve for ^{235}U interacting with thermal neutrons.

The major points of interest in this graph are summarized as follows:

Unsymmetrical yield. One fission fragment is almost always larger than the other, by a ratio of about 3:2.

The mass numbers of the fission products range from about 70 to 160.

Maximum yields of about 9% occur for mass numbers of 95 and 135. (Yields are independent of whether the neutrons are thermal or 14 MeV, as would come from a D-T source.)

FISSION NEUTRONS

As described above, neutron energy distribution is an important parameter in a reactor. As neutrons are created in fission, their most probable energy is about 0.5 MeV, with the average energy near 2.0 MeV. Neutrons are produced with energies in the MeV range (Figure 24-3-7), but the probability for fission is greatest with energies

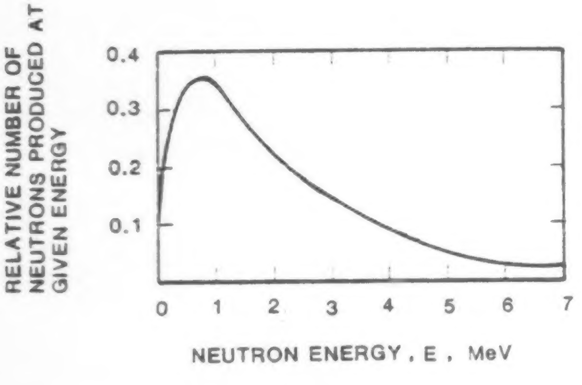


Figure 24-3-7.
Neutron energy spectrum covered by fission.

in the eV range (Figure 24-3-3); therefore, for most efficient operation, the neutrons must be slowed down or reduced in energy.

The process of slowing down the neutrons is called "moderation." A comparison of Figure 24-3-7 with 24-3-3 shows why moderation is necessary for efficient neutron use.

The average number of neutrons produced, $\bar{\nu}$, per event depends upon the fissile material and is summarized in Table 24-3-1.

TABLE 24-3-1

NUMBER OF NEUTRONS PRODUCED PER FISSION

Nuclide	$\bar{\nu}$
^{235}U	2.44
^{239}Pu	2.90
^{233}U	2.50

FISSION-PRODUCED ENERGY

The last, and perhaps most useful and important, fission product to be discussed is energy. To understand the source of this energy release, it is necessary to realize the interrelationship equating mass, m, and energy, E.

Energy is produced in the fission process by transformation of a minute amount of mass into energy according to Einstein's equation:

$$E = mc^2 \qquad \text{Eq. 24-3-4}$$

where

E = energy, ergs.
m = mass loss or gain, grams.
c = velocity of light ~3×10^{10} cm/s.

EXAMPLE

Energy released from typical fission reaction fission fragements:

^{97}Sr (at. mass = 96.923 amu).
^{136}I (at. mass = 135.907 amu).

No. of neutrons released = 3.

Reactants:

neutron		1.009	amu
^{235}U	+	235.044	amu
		236.053	amu (total mass of reactants).

Products:

^{97}Sr	96.923	amu
^{136}I	135.907	amu
3 neutrons	3.027	amu
	235.857	amu (total mass of products).

Difference in atomic mass units:

Reactants - products: 0.196 amu;

therefore; m = 0.196 in the equation $E = mc^2$ and

$E = m x \ (931.2 \ \frac{\text{MeV}}{\text{amu}})$

Energy released = 0.196 amu x (931.2 MeV/amu).

E = 182 MeV per fission process.

The exact energy per fission depends on the specific products selected. The rounded-off number most often used is 200 MeV per fission; therefore, this number is used in the following calculations.

Given that power has units of energy per time, it is now possible to calculate the fission rate necessary to generate a certain power.

EXAMPLE

Given: Power = 10 MW.

Useful constants:

Energy ~ 200 MeV/fission.

1 MeV = 1.6×10^{-6} erg.

1 erg/s = 10^{-7} watt.

Find: Fission rate (fissions/second) necessary to produce 10 MW of thermal power.

Solution:

Step 1. Convert 10 MW to erg/s:

$$10 \text{ MW} = 10 \text{ MW} \times \left(\frac{10^6 \text{ watts}}{\text{MW}}\right) \times \left(\frac{1 \text{ erg/s}}{10^{-7} \text{ watt}}\right)$$

$$= 10^{14}\ \frac{\text{erg}}{\text{s}}.$$

Step 2. Convert ergs/s to MeV/s:

$$10^{14}\ \frac{\text{erg}}{\text{s}} \times \left(\frac{1\ \text{MeV}}{1.6 \times 10^{-6}\ \text{erg}}\right)$$

$$= 6.25 \times 10^{19}\ \frac{\text{MeV}}{\text{s}}.$$

Step 3. Convert $\frac{\text{MeV}}{\text{s}}$ to fissions/s:

$$6.25 \times 10^{19}\ \frac{\text{MeV}}{\text{s}} \times \left(\frac{1\ \text{fission}}{200\ \text{MeV}}\right)$$

$$= 3.125 \times 10^{17}\ \frac{\text{fissions}}{\text{second}}.$$

Thus, the generation of 10 MW of thermal power requires 3.12×10^{17} fissions per second.

ORIGINS OF FISSION ENERGY

It has been shown above that the ultimate source of the energy is the mass difference between the products and the reactants. A point now to be considered pertains to how and when, during the fission reaction, is energy released, and to what extent is it a recoverable heat source. About 84% of the energy from the total fission process is kinetic (usable) energy of the fission fragments. Another 6% of a 200-MeV/fission occurs simultaneously with the fission process. It is in the form of kinetic energy of the neutrons and gamma rays emitted during fission. Both of these are also slowed down by the moderator and/or structural materials surrounding the reactor, and this energy is also usable as heat. The remaining 10% of the 200-MeV/fission is released after fission by radioactive decay of the fission products, generally during the β^- decay of the fission product chains. The majority of β^- particles and γ rays emitted during decay are captured within the core, and this energy too is recoverable as heat. Approximately 12 MeV (6%) per fission is emitted as neutrinos during the decay. Because of their low mass, neutrinos are not stopped within the reactor core and their energy is consequently lost as useful heat. A small source of energy results when neutrons, which do not cause fission, are captured via the (n,γ) reaction. The gamma rays released from these reactions, like the prompt fission γ rays, are stopped within the reactor core. Table 24-3-2 summarizes energy emitted and recovered per fission of ^{235}U.

TABLE 24-3-2

EMITTED AND RECOVERABLE ENERGY FROM FISSION OF A ^{235}U ATOM

	Emitted Energy (MeV)	Recoverable Energy (MeV)
Fission fragments	168	168
Fission neutrons	5	5
Prompt γ radiation	7	7
Fission product decay		
β radiation	8	8
γ radiation	7	7
Neutrinois	12	0
Capture γ radiation	0	~5
Total	207	~200

The fact that the fission products deliver up to 10% of the total energy per fission at a time well after fission shows why the power of a reactor does not drop instantly to zero the moment the chain reaction ceases. This 10% heat release occurs after the primary fission process and contributes to post-shutdown or decay heating. Because the fission products are radioactive, the rate of heating diminishes with time and is inversely proportional to the half-life of the average fission product.

A typical decay heating curve (Figure 24-3-8) assumes

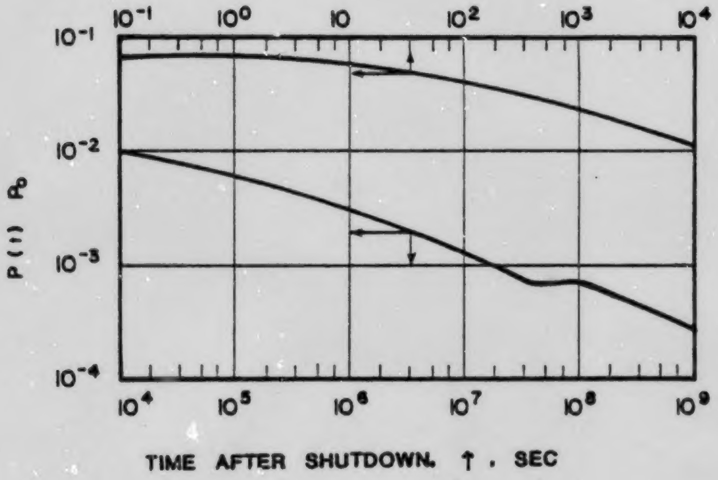


Figure 24-3-8. Ratio $P(t)/P_0$ of the decay heating to reactor operating power as a function of time, t, after shutdown.

the reactor has been operating long enough to build up an equilibrium concentration of the fission products. The time axis spans 10^{-1} second to 10^{9} seconds (about 30 years).

The vertical (power) axis is given as the ratio of the power at time t, P(t), to the full operating power, P_o.

A mathematical approximation of this curve is given by Equation 24-3-5, where t is expressed in days after shutdown following reactor operation of several months:

$$\frac{P(t)}{P_o} = (6.1 \times 10^{-3})\ (t^{-0.2}). \qquad \text{Eq. 24-3-5}$$

For example, the calculation of how much power remains in a 10-MW reactor one day after shutdown can be determined from either Figure 24-3-4 or Equation 24-3-5. From Figure 24-3-4, one day is 8.6×10^4 seconds and P(one day)/P_o = 5×10^{-3}. Thus, one day after shutdown 10 MW x 5×10^{-3} = 50×10^{-3} MW, or 50 kW of power remains. From Equation 24-3-5, P(one day)/P_o = $6.1 \times 10^{-3}\ (1)^{-0.2} = 6.1 \times 10^{-3}$ and the power after one day is computed to be 61 kW. Of these two answers, the one taken from the graph is probably the more accurate because one simple estimating equation cannot closely follow the entire curve.

SPONTANEOUS FISSION

So far, this module has discussed only particle-induced fission, for example, thermal neutrons interacting with ^{235}U. The student should be aware, however, that some heavy nuclides occasionally undergo spontaneous fission (one of the modes of radioactive decay). For example, about 3.5 x 10^{-7}% of the time ^{235}U undergoes spontaneous fission. The rest of the time it decays. Because of its low probability of occurrence, spontaneous fission is not significant in this study of neutron induced fission.

EXERCISES

1. Estimate the power of a 1000-MW reactor at the following times after shutdown of a very long run:
   - 1 hour.
   - 1 day.
   - 1 week.
   - 1 month.
   - 1 year.
   - 10 years.
2. Calculate the fissions per second to yield 1 watt.
3. Calculate the matter converted into energy via fission per second to yield 1 watt.
4. Write two fission reactions that may occur in a uranium reactor that has been operational for a while. (Use FF1 and FF2 to symbolize the two fission fragments.)

MATERIALS

Slide rule or hand calculator.
Handbook of Chemistry and Physics.
Sealed uranium source.
Energy-calibrated multichannel analyzer and Ge(Li) detector.
Source of neutrons.
G.E. Chart of the Nuclides.
Semilog paper.

PROCEDURES

1. Calculate mass of all fission products present in a 1000-MW (thermal) reactor after it has run continuously for one year:
   a. Calculate the seconds in a year __________(1).
   b. From second EXAMPLE, calculate fission per second required to give 1000-MW thermal ________(2).
   c. In one year __________(2) fission/s x __________(1) s/year = __________(3) fission/year.
   d. For each fission, one heavy and one light mass fission product will be formed. From Figure 24-3-6, determine most probable mass for the heavy and light product: $[\frac{\text{amu}}{\text{atom}} = \frac{\text{grams}}{\text{moles}}]$:

      __________(4) g/mole = M(heavy);
      __________(5) g/mole = M(light).

e. Calculate mass of the light mass products:

__________(3) fission/year x ____________(5)
g/mole x 1 atom/fission =
__________________________ =
6.023×10^{23} atom/mole
___________(6) g/year of light mass product.

f. Similarly to "e" above, calculate mass of the heavy mass product:

___________(3) fission/year x __________(4)
g/mole x 1 atom/fission
= _______________________ = ___________(7) g/year
6.023×10^{23} atom/mole
of the heavy mass product.

g. Total mass/year: ___________(6) + ___________(7)
= __________g/year.

2. Calculate volume of the fission product whose mass was calculated in Procedure 1 above.
For each fragment mass chosen in Procedure 1 above, and using the Handbook of Chemistry and Physics and the G.E. Chart of the Nuclides, fill in Data Table 24-3-1.

Note: Realize that the above calculations for Procedures 1 and 2 have assumed that all the fission fragments with compounds have densities of (7) g/cm^3, etc. Obviously, this is a generalized solution to the problem. To solve problems 1 and 2 exactly requires treating each mass separately, knowing exactly what compounds will be created at the reprocessing plant and then adding the weights and volumes for each element and compound to get a more exact solution for 1 and 2 respectively.

DATA TABLE 24-3-1

Description	Heavy Mass Product	Light Mass Product
1. Mass of fragment	________(d) gram/mole	________(e) gram/mole
2. The longest-lived radionuclide in the β^--decay chain with the mass in 1 above	________	________
3. A common compound of the element of 2 above; such a compound would be an oxide or sulfide	________	________
4. The molecular weight of the compound of 3 above	________	________ gram/mole
5. The mass fraction of element 2 above in compound 3 above	____atoms molecule^{-1}, ____(d)g mole^{-1} ____________________ ________(4) g mole^{-1} = ________	____atoms molecule $^{-1}$, ____(d) g mole^{-1} ____________________ ________(4) g mole^{-1} = ________
6. The mass per year of the fission fragment compound	________(g) g year^{-1} ____________________ ________(5) = ________ g year^{-1}	________(f) g year^{-1} ____________________ ________(5) = ________ g year^{-1}
7. Density of compound 3 above	________ g cm^{-3}	________ g cm^{-3}
8. The volume per year associated with each fragment	________(6) g year^{-1} ____________________ ________(7) g cm^{-3} = ________cm^3 year^{-1}	________(6) g year^{-1} ____________________ ________ g cm^{-3} = ________cm^3 year^{-1}
9. Total volume per year = ________(8) cm^{3}year^{-1}(Heavy + ________(8) cm^{3}year^{-1}(Light) = ____________ cm^3 year		

REFERENCE MATERIALS

Bennet, D.J. *The Elements of Nuclear Power*. New York: John Wiley and Sons, 1973.

Grant, P.M.; Miller, G.E. and Rowland, F.S. *Gamma Transition Energies of the Major Fission Products*, Report No. 1971-1. Irvine, CA: Nuclear Reactor Facility at University of California, 1971.

LaMarsh, John R. *Introduction to Nuclear Engineering*. Reading, MA: Addison-Wesley Publishing Co., 1975.

TEST

1. Describe contributions of Fermi, Hahn and Strassman in the discovery of fission:

2. Write a generalized nuclear equation for fission of ^{235}U:

3. Differentiate between fissionable, fissile and fertile materials:

4. List three most common fissionable nuclides:

5. Write nuclear equations for converting ^{238}U into ^{239}Pu, and ^{232}Th into ^{233}U:

6. Sketch and briefly explain curve of fission cross section versus neutron energy for ^{235}U:

7. Briefly explain why most fission fragments are β^- emitters:

8. Sketch and briefly explain fission yield curve for ^{235}U:

9. Calculate energy released in fission (given identity and masses of reactants and products):

10. Briefly explain why energy of a neutrino is not usable in a power reactor:

11. Explain distribution of fission energy between three major fission products:

12. Verify that 3.1×10^{10} fissions/second produces 1 watt:

13. (Given that 280 atoms of ^{235}U undergo spontaneous fission per second per ton of material), calculate number of tons of material required to produce one watt of power from spontaneous fission:

14. Calculate power [P(t)] 10 days after shutdown if full operating power (P_o) was 100 MW; use $\frac{P(t)}{P_o} = (6.1 \times 10^{-3}) \times (t^{-0.2})$:

15. Define the following terms:
    a. Transuranium elements.

    b. Spontaneous fission.

    c. Fission yield.

    d. Unsymmetrical yield.

    e. Decay heating.

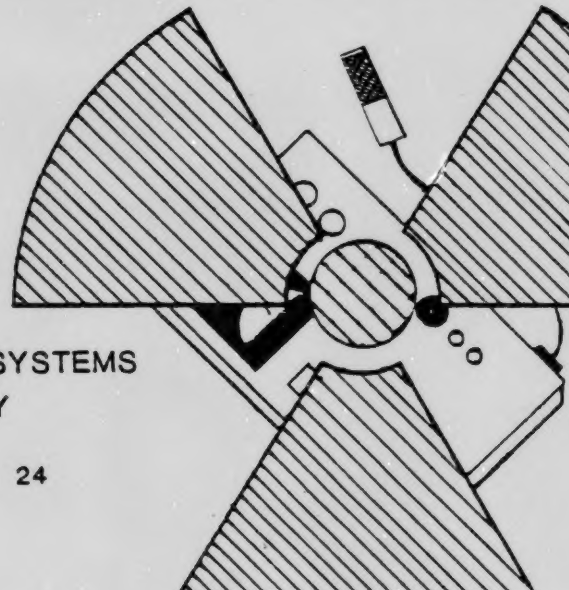
    f. Fissile.

    g. Fertile.

NUCLEAR TECHNOLOGY

MODULE 24-4

NUCLEAR CRITICALITY CONTROL



NUCLEAR SYSTEMS
AND SAFETY

COURSE 24

CENTER FOR OCCUPATIONAL RESEARCH AND DEVELOPMENT

COURSE MODULES

COURSE 24 NUCLEAR SYSTEMS AND SAFETY

MODULE 24-1	Neutron Physics
MODULE 24-2	Cross Sections and Reactions
MODULE 24-3	Fission Rate and Reactor Power
⇨ MODULE 24-4	Nuclear Criticality (0530 ERDA Manual)
MODULE 24-5	Nuclear Material Inventory Management Techniques
MODULE 24-6	Nuclear Plant Safeguards

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INTRODUCTION

A "critical mass" is any mass of nuclear material capable of sustaining a nuclear chain reaction. Criticality can release tremendous energy in the form of kinetic energy and penetrating radiations. The kinetic energy can damage equipment and spread contamination; the radiations can be lethal. A nuclear reactor is designed to contain these energies and harness them to produce power. Outside a reactor, these energies are exceedingly dangerous. Nuclear criticality safety is the art of avoiding the accidental release of these energies – the act of avoiding an accidental nuclear chain reaction.

This module is an introduction to nuclear criticality safety. It discusses the physics of a nuclear chain reaction and introduces general nuclear criticality safety procedures and practices.

MODULE PREREQUISITES

The student should have completed Course 1, "Radiation Physics"; Course 11, "Radiation Detection and Measurement"; Course 17, "Radiation Protection II"; and Course 20, "Radiation Monitoring Techniques."

OBJECTIVES

Upon completion of this module, the student should be able to:

1. Define the following terms:
    - Nuclear fission.
    - Fissile material.
    - Cross section.
    - Critical mass.
    - Nuclear criticality.
    - Nuclear criticality safety.
    - Operator.
    - Plant.
    - Safety guides.
    - Safe mass.
    - Geometry controls.
    - Moderation.
2. List at least three sources of nuclear criticality safety information.
3. Describe the fissioning of the ^{235}U nucleus.
4. List the three interactions that can occur when a neutron interacts with a nucleus.
5. Describe the relationship of cross section to the energy of an incident particle, the type of reaction, and the characteristics of the material undergoing reaction.
6. List three fissionable materials.
7. List and describe the means of controlling fission and the means of increasing the efficiency of the fission process.
8. Describe typical administrative policies of a nuclear criticality safety program.
9. List specific procedures that should be included in nuclear criticality safety procedures for receiving, inspecting, processing, and storing fissile materials.
10. List three ways to control the mass and distribution of nuclear material, and discuss their application.

11. List and discuss some factors that may affect the single parameter limit for an individual unit.

12. Discuss the operational characteristics of a criticality alarm system.

SUBJECT MATTER

Fission is a special case of radioactivity in which two or more heavy particles, as well as a number of light particles, are produced. Fission occurs when a neutron causes a nucleus to break into two or more nuclei, with the probable emission of additional neutrons and gamma rays. Figure 24-4-1 depicts the fissioning of ^{235}U, the naturally-occurring uranium isotope in which fission can be produced most easily. When a neutron interacts with the ^{235}U nucleus, a compound ^{236}U nucleus is formed. The ^{236}U nucleus splits almost instantaneously into two smaller (but fairly large) nuclei (such as ^{131}I and ^{102}Mo). Three neutrons and gamma radiation are emitted. The splitting of the nucleus is called "nuclear fission." Material that can be fissioned is called "fissionable material."

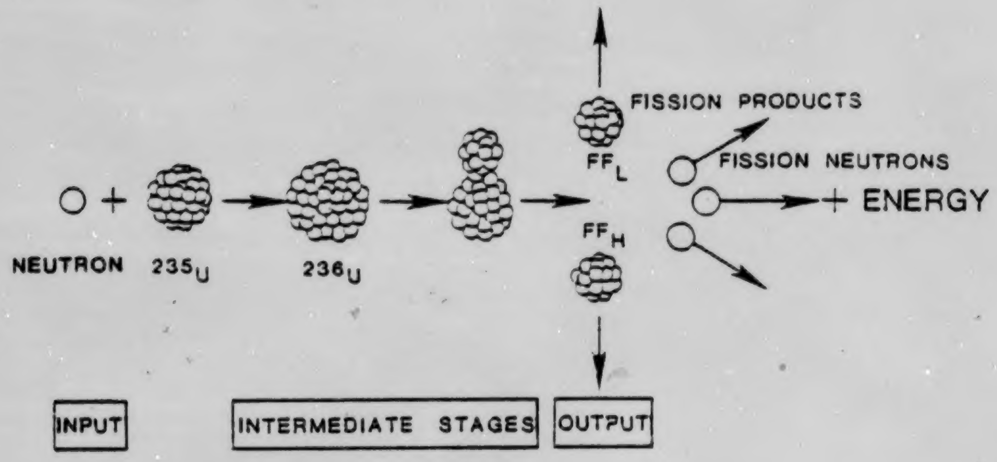
FISSION PRODUCTS
FISSION NEUTRONS
FF$_L$
+ ENERGY
FF$_H$
NEUTRON
^{235}U
^{236}U
INPUT
INTERMEDIATE STAGES
OUTPUT

Figure 24-4-1. Four stages in fission of ^{235}U. (FF = fission fragments; H and L = heavy and light.)

The fissioning of the nucleus is only one of three reactions that can occur when a neutron interacts with it. The neutron also could have been scattered or absorbed by the nucleus. Any of these reactions may occur, and the probability of their occurrence may be statistically determined. This probability is stated in terms of cross section.

CROSS SECTION

A cross section is a measure of the probability that a specific nuclear reaction may occur. It is expressed in units of area and may be thought of as expressing the size of the nuclear "target" with which the neutron interacts. It is a function of the type of reaction, the energy of the incident neutron, and the characteristics of the material undergoing reaction.

In general, cross section decreases as the energy of the incident particle increases. Slower neutrons (called "thermal neutrons" because they are in thermal equilibrium with their environment) are more likely to interact with the nucleus than faster neutrons. The cross section for radiative capture (absorption) of thermal neutrons in ^{235}U is about 107 barns (a barn is equal to 10^{-24} cm^2). The cross section for fission at thermal energy is about 580 barns. The statistical probability of fission, thus, is about six time that of capture.

FISSILE MATERIAL

^{235}U, ^{239}Pu, and ^{233}U are the three most abundant nuclides readily fissionable by thermal neutrons. Both ^{239}Pu and ^{233}U are fissile in a manner similar to ^{235}U, although some of their fundamental properties vary from nuclide to nuclide. Of the three, only ^{235}U is found in nature and has isotopic abundance of ~ 0.72 wt %.

THE NUCLEAR CHAIN REACTION

One or more neutrons are produced each time a ^{235}U nucleus splits. If one of these neutrons is captured by another ^{235}U nucleus, causing it to split, and if the neutron produced by that reaction causes another nucleus to split, and so on indefinitely, a nuclear chain reaction has been produced (Figure 24-4-2).

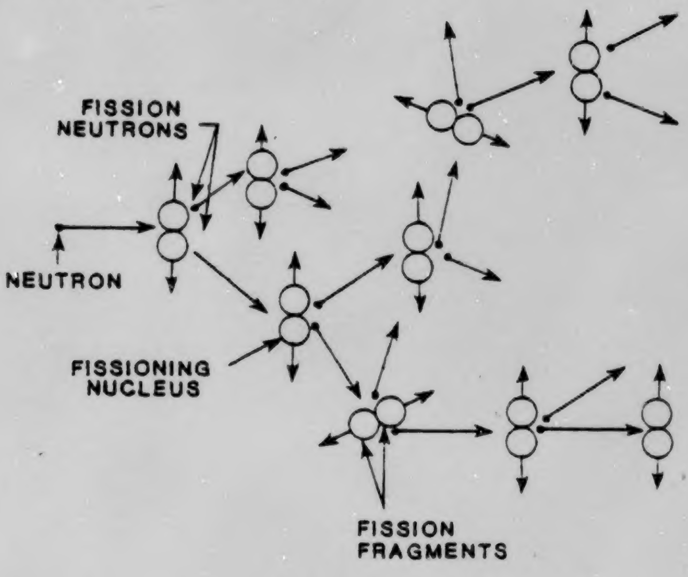

Figure 24-4-2. Fission chain reaction.

This chain reaction releases great amounts of energy as heat and radiation. A nuclear power reactor is constructed in such a manner as to control and use this energy. Outside the reactor, an inadvertent energy release through an accidental nuclear excursion obviously is to be avoided. The avoidance of a nuclear criticality accident is the heart of the nuclear criticality safety program.

CONTROL OF CRITICALITY

On the average, 2.43 neutrons are produced when a ^{235}U nucleus splits under bombardment with thermal neutrons. Only one of these neutrons is necessary to induce a succeeding fission in the system. The other neutrons either are absorbed by associated nonfissile materials or by ^{235}U to produce ^{236}U. Others may escape from the system entirely (leakage).

The production of fission neutrons depends upon the distribution, density, and quantity of fissile material. Absorption depends on the presence of neutron-absorbing material. Leakage depends on the shape of the fissile unit and the presence of surrounding materials that can return neutrons back into the unit by scattering process.

These properties suggest some means of fission control:

- Mass and volume control.
- Neutron absorption.
- Geometry control.
- Concentration control.

They also suggest some means by which the efficiency of the fission process can be increased, including –

- Neutron moderators.
- Neutron reflectors.

Each of these means of control is discussed specifically in the next section of this module and is related to actual nuclear criticality safety practice.

NUCLEAR CRITICALITY SAFETY PRACTICES

Fissile materials must be processed, stored, transferred, and handled in such a manner that the probability of an accidental chain reaction is at an acceptably low level. To this end, several documents have been drafted to regulate and guide nuclear criticality safety practices. The most important of these are Chapter 0530, "Nuclear Criticality Safety" contained in the Energy Research and Development Administration Manual (ERDA/DOE), the Nuclear Safety Guide TID-7016, Revision 2, sometimes referred to as the "Criticality Bible," and various national standards for nuclear criticality safety.

Chapter 0530 contains provisions and regulations that apply only to programs in DOE-owned plants and to activities that involve fissile materials and are performed under contract with DOE. These provisions do not apply to non-government related commercial facilities. The Nuclear Safety Guide, however, is applicable to both government and commercial facilities. It was prepared by a nuclear safety group in 1955 and has been revised since. It especially covers nuclear criticality practices in fuel reprocessing, fabrication, enrichment, transportation, and oxide conversion.

A number of American National Standards have been written for nuclear criticality safety. These standards are available from the American Nuclear Society and are listed in the reference section of this module.

The knowledge of the precise meaning of commonly used terms is essential to a discussion of nuclear criticality safety:

- Critical Mass. The mass of fissile material that will support a self-sustaining chain reaction under stated conditions.

- Fissile Materials. Isotopes capable of sustaining a neutron-induced fission chain reaction: ^{233}U, ^{235}U, ^{239}Pu, ^{238}Pu, and ^{241}Pu. (Additional isotopes may be included as significant quantities become available.)
- Nuclear Criticality. The state in which the effective neutron multiplication constant (K_{eff}) of a system of fissile material equals or exceeds unity; that is, a self-sustaining chain reaction.
- Nuclear Criticality Safety. Prevention of, or protection against, injury or damage due to an accidental nuclear criticality.
- Operator. Any individual who manipulates or handles fissile materials or manipulates the controls of equipment used to produce, process, store, or package such materials.
- Plant. Any facility or system, except a nuclear reactor or critical assembly (as defined in chapter 0540), wherein fissile materials are produced, stored, processed, transferred, or handled in quantities or combinations of quantities and under conditions that could result in accidental criticality.
- Safety Guides. Documents designated or recognized as an acceptable basis for nuclear criticality safety practices. The guides are used as aids by DOE field office managers when suggesting acceptable safety practices, and include material developed by DOE contractors, professional societies, industrial organizations, and foreign atomic energy industries.
- Safe Mass. That mass of fissile materials that is subcritical for all credible conditions to which it could be exposed, including processing, handling, and procedural uncertainties.

THE PROGRAM

Several factors must be considered when a nuclear criticality safety program is set up. In general, these considerations can be classified into two groups:

- Factors relating to the organization and administration of the program.
- Factors relating directly to the physics of nuclear criticality — the means by which a nuclear chain reaction can be prevented.

These factors are inevitably intertwined. A consideration of one often demands or depends upon the consideration of the other. They are separated here in order that each of them can be discussed individually, but it must not be forgotten that each of them is only a part of the total program.

ADMINISTRATIVE POLICIES

Among the administrative and organizational considerations are —

- The selection and training of personnel to operate the plants and administer the safety policies.
- The writing of plans and procedures to govern actions involving fissile materials.
- The provision of a system for review and audit of the safety program.

In a general way, Chapter 0530 establishes guidelines for the selection, training, and examination of persons who operate or supervise plants and stipulates that operators and supervisors possess formal training and/or pertinent experience for their position and be physically capable of carrying out their work. The chapter mandates a formal training and retraining program to be conducted during company time. Formal examinations must be conducted. It further specifies that a program be provided to train and retrain operators and supervisors to the extent necessary to promote an awareness of all nuclear criticality safety requirements.

Written plans and procedures that govern all actions that involve fissile material must be incorporated in the safety program. These procedures must include plans for the receiving, inspecting, storing, and processing of fissile materials. Tables 24-4-1 through 24-4-3 list, under each of these categories, some of the details that must be considered in the preparation of these written plans.

TABLE 24-4-1. RECEIVING AND INSPECTING FISSILE MATERIAL.

Plans for receiving fissionable material into the facility and for inspecting the shipment on receipt, must include procedures for –

- Determination, verification, or noting the contents of each package, including the net weight of fissionable material therein.
- The placing of material in the receiving area and the storage facility.
- Handling of wet and/or damaged packages.

TABLE 24-4-2. STORAGE OF FISSILE MATERIALS.

Plans and procedures for storage of fissile material must include —

- Limitations and total quantity of material, quantity limit for each individual unit, dimensions of containers, and spacing between units.
- Description of containers in which fissile materials are stored.
- Description of the storage facility, including dimensions and materials of construction of the enclosure and shelving, cubicles, cages, and other equipment within the storage area.
- Precautions to avoid entry of water or other moderating material into a storage area in which moderating and reflecting effects could compromise nuclear safety.
- Administrative controls over the distribution of fissile material from storage and its return to storage, including means of verifying the weight, isotopic content, chemical composition, and degree of moderation.

TABLE 24-4-3. PROCESSING FISSILE MATERIAL.

Plans and procedures for processing the fissile material must include —

- A description, using appropriate sketches or drawings of equipment and facilities in which the potential for criticality exists, showing dimensions in sufficient detail to permit evaluation of the information mentioned below (the last four items).

- A statement of the chemical and physical form of fissile material in each step of the process, including isotopic content, the nature of any material, and the resulting concentrations, densities, and degrees of moderation throughout the steps of the process.
- A statement of the maximum quantities of fissile material at any one time in each step of the process.
- Spacing of masses of fissile material within each process area and separation from fissile material in adjoining areas.
- Methods of collecting, handling, and transporting products from each process area of individual operation and evaluation of the nuclear safety of these methods.
- Identification and description of procedures necessary to prevent criticality resulting from accumulation of fissile material in scrap or waste, lathe turnings, crucible slag, pickling solutions, choppings, sumps, filters, etc.
- Description of a criticality alarm system and emergency procedures.

Chapter 0530 also establishes procedures for safety review by competent technical personnel other than those having direct responsibility for the operation. Essentially, this is an internal safety review system that can be reviewed by DOE and contractor management. This system acts as a check and demonstrates compliance with safety procedures.

PHYSICAL CONSIDERATIONS

These considerations include the following:

- Control parameters for nuclear criticality safety.
- Factors affecting these control parameters.
- Instrumentation.

CONTROL PARAMETERS

Chapter 0530 sets down the basic control parameters for nuclear criticality safety. These are the controlling factors that can maintain a fissile mass in a subcritical state. Control can be achieved either by controlling the mass and distribution of all fissile material or by controlling the mass distribution and use of the nuclear properties of all other materials with which fissile elements are associated.

There are three methods for the control of the mass and distribution of fissile material:

- Mass and Volume Control.

  A certain quantity of fissile material is necessary for the production of a nuclear chain reaction. Similarly, the volume occupied by the fissile material must not be less than a certain value or leakage will be excessive. Chapter 0530 recognizes mass and volume control as a valid means of controlling criticality.

  > For operations where nuclear criticality safety depends upon mass control, the allowable mass shall be no greater than the safe mass for the associated conditions. The safe mass in all cases shall be based upon current published or available experimental determinations and/or published or

> available nuclear safety guides and handbooks. These guides and handbooks may include values which, in the absence of directly applicable experimental measurements, are derived from calculations made by a method shown to be valid by comparison with experimental data, provided allowances are made for uncertainties in the data and in the calculations. For operations depending upon mass control wherein the contained volume does not automatically limit the contents to the safe mass or less, the possibility of multiple batching shall be considered. If a batch of fissionable materials consists of different physical and chemical forms of a particular isotope, i.e., metallic U^{235}, compounds of U^{235}, etc., the safe mass for the most reactive combinations under the associated conditions shall be the governing criterion. If a batch of fissionable materials consists of a mixture of fissionable isotopes, i.e., Pu^{239}, U^{235}, and U^{233}, the allowable safe mass shall be determined experimentally or determined from calculations made by a method shown to be valid by comparison with experimental data.

- Density Control

If the material density of a fissile material is high, the quantity required for a chain reaction is, in general, less than that required if the nuclei were separated more widely. Density is an accepted control parameter.

> Density Controls. Density (mass of fissionable isotope per unit volume) is an accepted parameter applicable to control for nuclear criticality safety. Systems wherein density control is used shall meet established criteria for density considerations. These criteria may be found in guides, handbooks, and data compilations.

- Geometry Control.

Geometry can be controlled to increase leakage. Leakage can be increased by making the surface-to-volume ratio high. Long, thin pipes are better than spheres. Subcriticality can be assured by limiting of diameters or thicknesses to safe dimensions. Controls of this type, called "geometry controls," are built into the system and are considered more reliable than mass or volume controls since they are, in a sense, automatic. Pipes and vats are constructed in such a way that a safe geometry is automatically insured:

> Spacing Controls. Individual items of equipment and/or containers holding fissionable materials, when arranged in a group, in storage, or when being transferred within a plant or between facilities onsite, shall be spaced so that the entire array is subcritical for all conditions that affect or might affect the plant, facility, or site. Movement of material under credible in-plant and on-site accident conditions shall be considered.

Criticality safety may be achieved by the use of neutron absorbers, moderation controls, and neutron reflector control:

- Neutron Absorbers.

The presence of nuclei or other elements that have relatively high neutron absorption cross sections will require an increase in the quantity of fissile material necessary for a chain reaction. Neutron absorbers can be mixed with the fissile material or be a part of the equipment structure because absorbers will absorb neutrons and help to maintain subcriticality.

> Neutron Absorbers. Neutron-absorbing materials, such as cadmium and boron, may be used to make equipment and processes safe, provided available data confirm their suitability and assure their presence and reliability. Care should be exercised in the use of solutions of neutron absorbers because of the controls required to assure their continued effectiveness.

- Control of Neutron Moderation.

Light nuclei, such as hydrogen, can reduce the speeds of neutrons at which they collide to thermal levels. This reduction increases the possibility of a chain reaction. The reduction of energy, or slowing down of neutrons, is called "moderation." The material with which the neutron interacts to lose energy is termed the "moderator." The presence of moderating material, along with fissile nuclei, increases the efficiency of the fission process. Materials that contain hydrogen – such as water, oil, many plastics and hydrocarbons – are excellent moderators. When they are mixed with fissile materials, their distribution and concentration must be carefully controlled:

> Moderation Controls. For operations where nuclear criticality safety depends upon control of neutron moderation, there shall be assurance that the prescribed extent of moderation remains unchanged or if it does change, the reactivity of the system remains below acceptable subcritical limits. Such assurance shall include consideration of all credible accidents.

- Neutron Refelection Control.

Materials may act as neutron reflectors and allow neutrons to "bounce" off them and give them another opportunity to produce fission. If the reflector has a low atomic number, its effectiveness is increased because the neutrons are not only returned but may be returned at a more effective energy. Neutron reflection of a unit causes a decrease in leakage and an increase in moderation. Although any material can be a reflector, the materials most usually encountered are water, concrete, plastics, the common materials of construction, and people:

> Neutron Reflection. Neutron reflection shall be considered for all systems of fissionable material. The extent of reflection shall be based upon the actual reflectors present or those to be expected during normal operations or as a result of a credible accident.

A more technical discussion of these parameters is provided by American National Standard (ANS) N16.1 "Limits for Individual Units" and Chapter II of the Nuclear Safety Guide. These references establish and discuss single-parameter limits for fissile nuclides. These limits apply when only a single parameter, such as a mass of fissile material, is controlled to prevent criticality. These limits are based upon experimental data.

Parameter limits are provided for hydrogen-moderated systems, for systems that rely upon concentration limits to maintain subcriticality, for mixtures of nuclides, and for special geometries.

Tables 24-4-4 and 24-4-5, taken from the Nuclear Safety Guide, are typical and indicate the magnitude of the basic nuclear parameters.

TABLE 24-4-4. SINGLE-PARAMETER LIMITS FOR UNIFORM AQUEOUS SOLUTIONS REFLECTED BY AN EFFECTIVELY INFINITE THICKNESS OF WATER*.

Parameter	Subcritical limit for		
	^{235}U	^{233}U	^{239}Pu N:Pu > 4
Mass of fissile nuclide, kg	0.76	0.55	0.51
Solution cylinder diameter, cm	13.9	11.5	15.7
Solution slab thickness, cm	4.6	3.0	5.8
Solution volume, liters	5.8	3.5	7.7
Concentration of fissile nuclide, g/liter	11.5	10.8	7.0
Areal density of fissile nuclide, g/cm^2	0.40	0.35	0.25
*These values are from the fifth Reference.			

TABLE 24-4-5. THICKNESS OF REFLECTORS REQUIRED FOR THE CRITICALITY OF A SPHERE OF EACH OF VARIOUS FISSILE MATERIALS.

Fissile Material		Reflector Material Thickness (cm)					
Form and density	Radius (cm)	Water (1.0 g/cm^3)	Iron (7.86 g/cm^3)	D$_2$O (1.10 g/cm^3)	Carbon (1.90 g/cm^3)	Beryllium (1.80 g/cm^3)	Plexiglas[a] (1.20 g/cm^3)
^{235}U Metal (18.82 g/cm^3)	6.46	15	17.56	7.23	8.36	3.76	5.05
^{239}Pu Metal (19.85 g/cm^3)	4.00	15	16.21	7.64	8.07	3.20	6.05
^{235}U-Water (50 g/liter)	15.68	15	10.52	10.32	7.68	4.07	6.07
^{239}Pu-Water (30 g/liter)	15.71	15	8.99	9.64	7.21	3.86	5.87

[a] Methacrylate plastic, $C_5H_8O_2$.

FACTORS THAT AFFECT CONTROL PARAMETERS

The single parameter limit for an individual unit can be affected by several factors. Examples include –

- A change in the intended shape or dimensions, as a result of bulging or bursting of a container, of corrosion, or of failure to meet specifications in fabrication.
- An increase in the intended mass of fissile material as the result of operational error.
- A change in the intended ratio of moderator to fissile material resulting from –
    - Inaccuracies in instruments or in chemical analyses.
    - Loss of moderator by evaporation or displacement.
    - Addition of moderator to concentrated solutions.
    - Precipitation of fissile materials from solutions.
- A change in the effectiveness of neutron absorbers resulting from –
    - Loss of solid absorber by corrosion.
    - Loss of moderator.
    - Redistribution of neutron absorber and fissile material by precipitation of one, other,
    - Redistribution by clumping of solid neutron absorber within a matrix of moderator or solution.
    - Failure to add the intended amount of neutron absorber to a solution or failure to add it with the intended distribution.
    - Loss or decrease in concentration of neutron absorber through process operation.

Many of these occurrences are anticipated by the Nuclear Safety Guide and are discussed in Chapter III. Tables are provided to allow the technician to determine factors by which subcritical mass limits must be increased or decreased as a result of these occurrences. Figure 24-4-3, for example, gives the factors by which the subcritical mass limit for fissile metals may be increased when the metals are diluted by nonmoderating elements. Similarly, Figure 24-4-4 shows factors

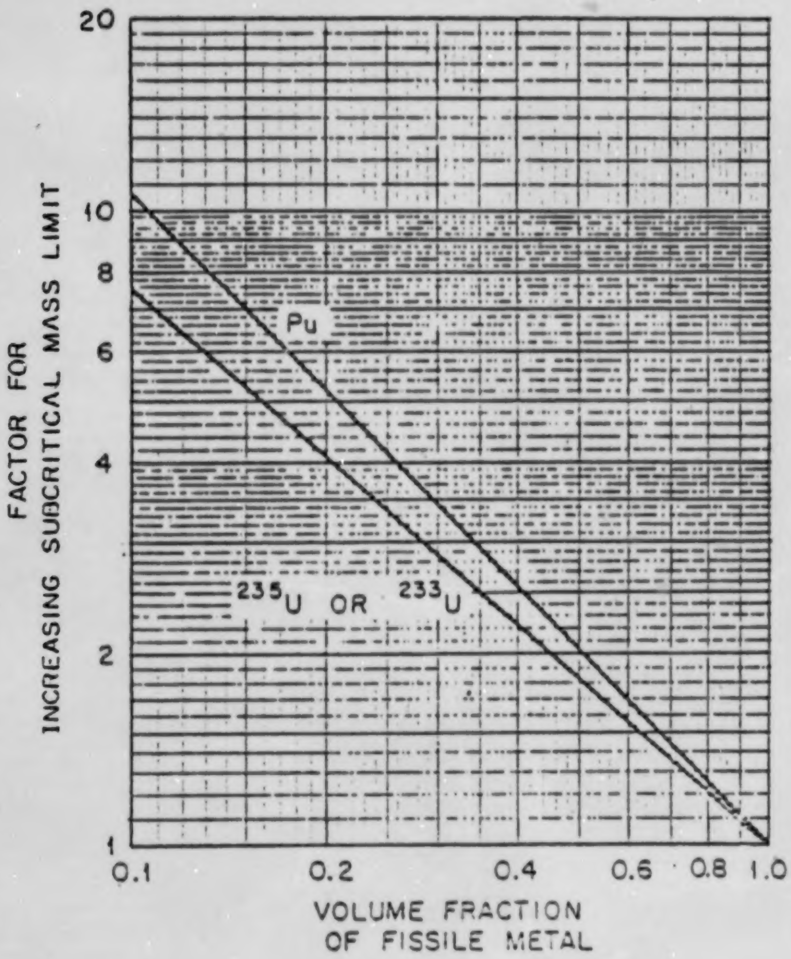

Figure 24-4-3. Factors by which subcritical mass limits for fissile metals may be increased as a result of dilution by nonmoderating elements $11 \leq Z \leq 83$.

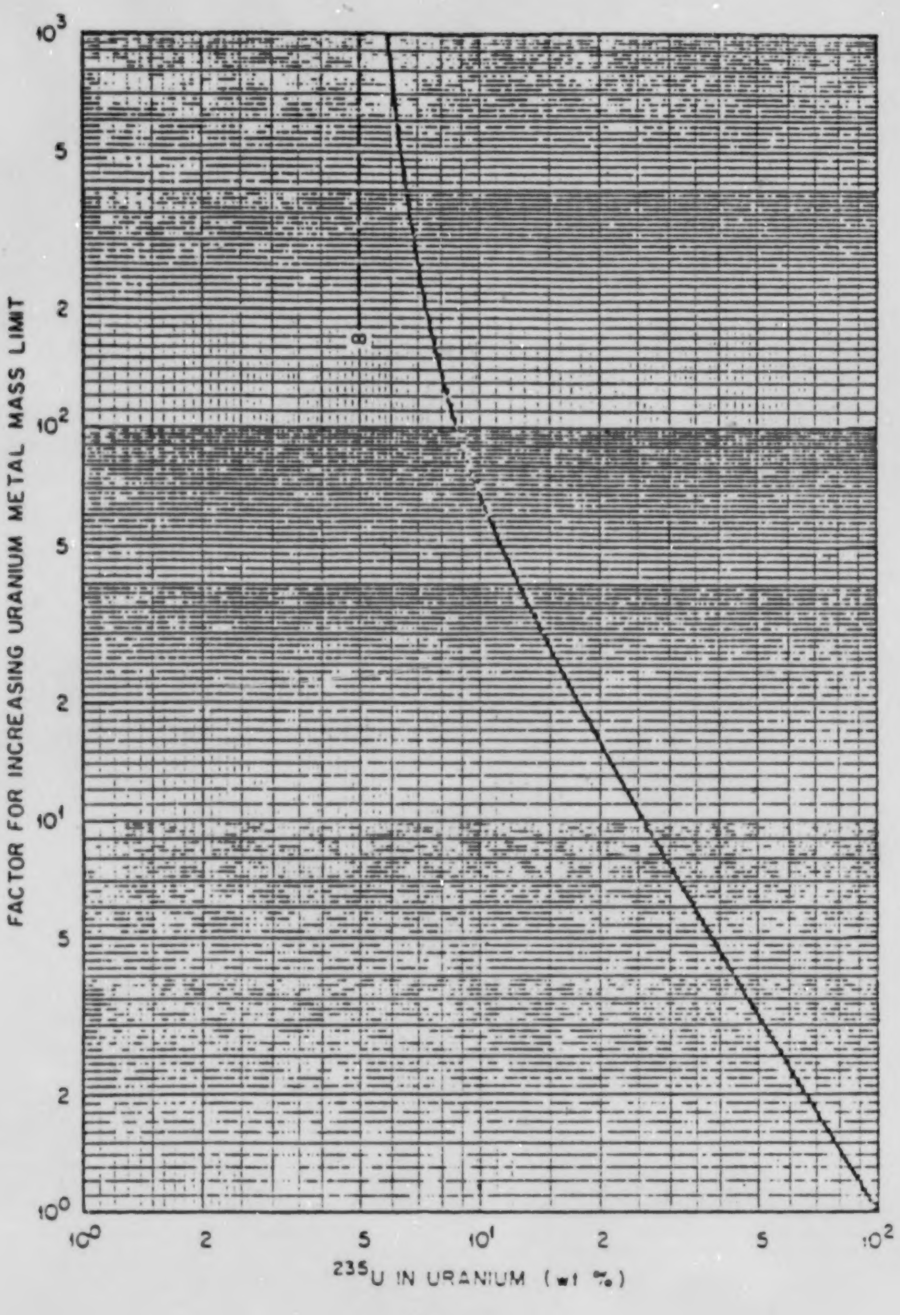

Figure 24-4-4. Factors by which ^{235}U metal subcritical mass limits may be increased for reduced uranium enrichment.

by which ^{235}U metal subcritical mass limits may be increased for reduced uranium enrichment. Many of these tables may be found in the American National Standards works listed in the Reference Materials section of this module.

INSTRUMENTATION

Criticality alarm systems are required by Chapter 0530. These systems are gamma- or neutron-sensitive radiation detectors that energize alarm signals if accidental criticality occurs. Unfortunately, these systems warn after the accident has occurred. Conditions in plants rarely allow detection of accidental approaches to criticality in order that conditions might be corrected before the radiation level becomes too dangerous. The warning is inevitably too late, except to signal prompt evacuation. These systems must not be relied upon to signal an approaching accident.

THE NUCLEAR CRITICALITY SAFETY RECORD

Since 1958, there have been seven supercritical accidents in chemical process equipment and none associated with mechanical storage, processing, or transportation. All occurred with aqueous solutions: four involved highly enriched uranium and three involved plutonium. Each accident was related to difficulties with equipment or with procedural inadequacies or violations.

Five of the accidents occurred between 1958 and 1962, a period when plants designed for moderate capacity were called upon to handle a greatly increased throughput and a variety of operations. As a result, the accident potential increased.

The influence of this cluster of accidents was pronounced. Criticality safety was refined, and the accident record was improved vastly.

EXERCISES

Use appropriate tables to complete the following exercises.

1. What thickness of reflector is required for the criticality of a sphere of each of the following materials?
   - A sphere of ^{235}U metal with a radius of 6.46 cm and a density of 18.82 g/cm^3. Water is the reflecting material.
   - A sphere of ^{239}Pu metal with a radius of 4.00 cm and a density of 18.85 g/cm^3. Beryllium is the reflecting material.

2. What is the mass single-parameter limit for a uniform aqueous solution of ^{235}U reflected by an effectively infinite thickness of water?

3. What is the concentration single-parameter limit of ^{233}U in a uniform aqueous solution reflected by an effectively infinite thickness of water?

4. By approximately what factor may the subcritical mass limit for ^{235}U be increased as a result of a dilution by nonmoderating elements to a 0.2 volume fraction of fissile metal?

MATERIALS

None.

PROCEDURES

None: Because of the potential hazard involved, a laboratory exercise is not included in this module.

DATA TABLES

None.

REFERENCE MATERIALS

American National Standard Criteria for Nuclear Criticality Safety Controls in Operations where Shielding Protects Personnel, N16.8. La Grange Park, IL: American Nuclear Society, 1975.

American National Standard Criticality Accident Alarm System, N16.2. La Grange Park, IL: American Nuclear Society, 1969.

American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors, N16.1. La Grange Park, IL: American Nuclear Society, 1975.

American National Standard for Safety in Conducting Subcritical Neutron-Multiplication Measurements IN SITU, N16.3. La Grange Park, IL: American Nuclear Society, 1975.

American National Standard Immediate Evacuation Signal for Use in Industrial Installations Where Radiation Exposure May Occur, N2.3. New York: American National Standards Institute, 1967.

American National Standard Safety Guide for the Performance of Critical Experiments, N405. La Grange Park, IL: American Nuclear Society, 1975.

American National Standard Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material, N16.4. La Grange Park, IL: American Nuclear Society, 1971.

American National Standard Validation of Calculational Methods for Nuclear Criticality Safety, N16.9. La Grange Park, IL: American Nuclear Society, 1975.

Code of Federal Regulations-Energy, Title 10, Chapter 1, Part 71.

Nuclear Safety Guide TID-7016. Revision 2. Prepared for the U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research, Oak Ridge, TN, 1978.

Operational Accidents and Radiation Exposure Experience Within the United States Atomic Energy Commission 1943-1970, WASH-1192. Washington, DC: U.S. Atomic Energy Commission, 1971.

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Paxton, H.C. *Criticality Control in Operations with Fissile Material*, LA-3366. Revised. Los Alamos, NM: Los Alamos Scientific Laboratory, 1972.

Proposed American National Standard Nuclear Criticality Safety Guide for Pipe Intersections Containing Aqueous Solutions of Enriched Uranyl Nitrate, N16.7. La Grange Park, IL: American Nuclear Society.

Stratton, W. R. *A Review of Criticality Accidents*, LA-3611. Los Alamos, NM: Los Alamos Scientific Laboratory, 1967.

U.S. Energy Research and Development Administration ERDA Manual, Chapter 0530. "Nuclear Criticality Safety" and Appendix. December, 1976.

TEST

Section I – Matching

1. Match the terms in column "A" with the appropriate definitions and descriptions in column "B."

________ Nuclear fission.

________ Fissile material.

________ Cross section.

________ Critical mass.

________ Nuclear criticality.

________ Nuclear criticality safety.

________ Operator.

________ Plant.

________ Safe mass.

________ Geometry controls.

________ Moderation.

________ Safety guides.

A. Reduction of the energy of neutrons.

B. Assuring subcriticality by limiting dimensions of equipment or containers holding fissile materials.

C. That mass of fissile materials subcritical for all credible conditions.

D. Documents designated as an acceptable basis for nuclear criticality safety practices.

E. Any facility or system, except a nuclear reactor or critical assembly wherein fissile materials are produced, stored, processed, or handled in quantities or combinations of quantities and under conditions that could result in accidental criticality.

F. That mass of fissile materials that is subcritical for all conditions to which it will be subjected during processing.

G. Any individual who manipulates or handles fissile material.

H. Prevention of or protection against injury or damage due to an accidental nuclear criticality.

I. The state in which the effective neutron multiplication constant of a system of fissile material equals or exceeds unity.

J. The mass of fissile materials that will support a self-sustaining chain reaction.

K. Isotopes capable of sustaining a neutron-induced fission chain reaction.

L. A measure of the probability that a specific nuclear reaction may occur.

M. A special care of radioactivity in which two or more heavy particles as well as a number of light particles are produced.

Section II — Multiple Choice

Choose the best response(s) to the questions below.

2. As the energy of the incident particle increases, cross section —
   a. Increases.
   b. Decreases.
   c. Remains the same.
3. The three most abundant nuclides readily fissionable by thermal neutrons are —
   a. ^{233}U, ^{239}Pu, ^{235}U.
   b. ^{241}Pu, ^{229}Th, ^{235}U.

c. ^{236}U, ^{235}U, ^{241}Pu.

d. ^{234}U, ^{233}U, ^{242}Pu.

4. Neutron reflectors may be used to –
   a. Increase the efficiency of the fission process.
   b. Control the fission process.
   c. Decrease the efficiency of the fission process.
   d. Absorb reflected neutrons.

5. A single parameter limit applies when –
   a. Systems are moderated by hydrogen.
   b. Controls are established by variation of the mass and geometry of fissile material.
   c. Only the mass of fissile material is controlled to prevent criticality.
   d. Mixture of fissile nuclides must be controlled.

6. Typical neutron absorbing materials include –
   a. Cadmium.
   b. Boron.
   c. Water.
   d. Concrete.

7. Criticality alarm systems warn –
   a. Only after the accident has occurred.
   b. Before the accident has occurred.
   c. Only of the approach of certain kinds of criticality accidents.
   d. None of these.

8. A change in the effectiveness of a neutron absorber may be caused by –
   a. Loss of moderator.
   b. Corrosion.
   c. Precipitation of fissile materials from solution.
   d. Addition of moderator.

Section III – Essay

9. List three sources of nuclear criticality information, and describe the types of information they provide.

10. Describe the fissioning of the ^{235}U nucleus.

11. List and describe the means of controlling fission and the means of increasing the efficiency of the fission process.

12. Describe typical administrative policies of a nuclear criticality safety program.

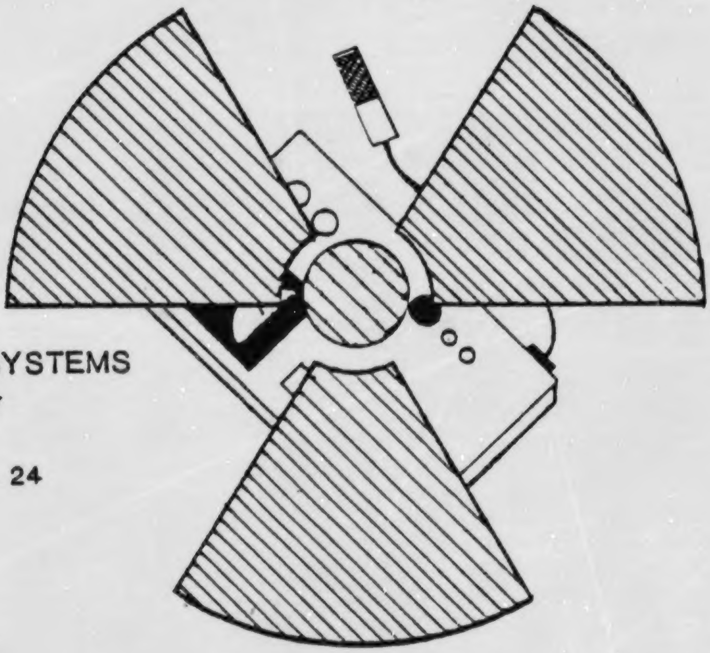
13. List and describe specific procedures that should be included in a nuclear criticality safety program for the receiving, inspecting, processing, and storing of fissile materials.

14. List three methods by which the mass and distribution of nuclear material can be controlled, and discuss their application.

NUCLEAR TECHNOLOGY

MODULE 24-5

NUCLEAR MATERIAL INVENTORY MANAGEMENT TECHNIQUES



NUCLEAR SYSTEMS
AND SAFETY

COURSE 24

CENTER FOR OCCUPATIONAL RESEARCH AND DEVELOPMENT

COURSE MODULES

COURSE 24 NUCLEAR SYSTEMS AND SAFETY

MODULE 24-1	Neutron Physics
MODULE 24-2	Cross Sections and Reactions
MODULE 24-3	Fission Rate and Reactor Power
MODULE 24-4	Nuclear Criticality (0530 ERDA Manual)
⇨ MODULE 24-5	Nuclear Material Inventory Management Techniques
MODULE 24-6	Nuclear Plant Safeguards

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INTRODUCTION

Nuclear materials – especially the fissile isotopes ^{233}U, ^{235}U, ^{239}Pu, and ^{241}Pu – are designated by Federal regulation as special nuclear materials and have great monetary and strategic value. Procedures have been established since 1947 to control and account for these materials. These procedures include an accountability system mandated by Federal regulation. This system is based upon the concept of the closed material balance.

This module explains the accountability system – what it is, how it works, and why it is important to nuclear material control.

MODULE PREREQUISITES

The student should have completed Course 1, "Radiation Physics"; Course 2, "Radiation Protection I"; Course 3, "Principles of Process Instrumentation"; Course 17, "Radiation Protection II"; and Course 23, "Nuclear Chemical Processes."

OBJECTIVES

Upon completion of this module, the student should be able to:

1. Define the following terms:
    - Nuclear material inventory management.
    - Source nuclear material.
    - Special nuclear material.
    - Strategic special nuclear material.
    - Material balance area.
    - Closed material balance.
    - Transfer journal.
    - Inventory ledger.
    - Summary ledger.
    - Internal records.
    - Moving and shutdown inventory.
2. State the equation for the closed material balance.
3. Describe inventory difference (ID); cite its significance to the closed material balance.
4. Describe the basic record system, citing its purpose and its components. Describe the transfer journal, inventory ledger and summary ledger; show how they are used to establish inventory difference.
5. List and describe records internal to a material balance area.
6. Describe the purpose and necessity of the physical inventory.
7. List and describe two physical inventory techniques and the advantages and/or disadvantages of each.
8. List the three steps basic to measurement of special nuclear material.
9. Describe techniques for measurement of weight and volume of special nuclear materials and for sampling of the materials.

10. Use statistical techniques to evaluate measurement accuracy and accuracy of material control information.

11. List three steps that should be followed to evaluate inventory difference.

SUBJECT MATTER

NUCLEAR MATERIAL INVENTORY MANAGEMENT

Nuclear material inventory management includes the performance of nuclear material inventory measurements, maintenance of written procedures (per 10 CFR 70.51c) and verifiable records, preparation of reports, and performance of data analysis to account for nuclear material. Nuclear material is meant to include all source nuclear material and all special nuclear material. Source nuclear material refers to thorium and depleted (in the isotope 235) or natural uranium. Special nuclear material (SNM) refers to plutonium, uranium-233, and the element uranium enriched in the isotope uranium-233 or in the isotope uranium-235. Strategic special nuclear material (SSNM) is a special category of SNM which includes plutonium, uranium-233, and the isotope uranium-235 (contained in uranium enriched to 20% or more in the uranium-235 isotope). A material balance area (MBA) is an identifiable physical area into (or out of) which only measured quantities of SNM are moved. An MBA also may be defined as "the smallest unit of any nuclear facility to which the principles of the closed material balance are applied."

Item control areas (ICA) are localities from which material (SNM) is identified by item as it enters and as it leaves the area. The validity of the quantity of items is ensured by tamper-safing devices.

THE CLOSED MATERIAL BALANCE

The closed material balance system is the system that establishes optimum accountability for SNM. It is a combination of accounting and engineering balance systems and deals with masses of materials. Closed material balance can be stated in the form of Equation 24-5-1.

$$\begin{array}{l}\textbf{Beginning Inventory +}\\ \textbf{Receipts - Shipments -}\\ \textbf{Discards or Losses -}\\ \textbf{Ending Inventory}\end{array} = \begin{array}{c}\textbf{Inventory Difference}\\ \textbf{(ID)}\end{array} \qquad \begin{array}{c}\text{Equation 24-5-1}\\ \text{(Equation = Eq.)}\end{array}$$

Beginning inventory for a given accounting period is the inventory at the end of the last accounting period. The terms "receipt" and "shipment" are self-explanatory. They are the known inputs (receipts) and known outputs (shipments, discards, or losses). In the case of a power reactor, however, the term "receipt" can acquire a special meaning and may be used in a special way. Thus, the production of plutonium in a reactor is considered a kind of receipt; therefore, the comsumption of ^{235}U is considered a kind of discard.

All entries in a closed material balance system must be measured directly. These measurements are subject to uncertainties (discussed later in the module); but in an ideal situation, the equation will balance. The equation balances when ID is close enough to zero for it to be statistically possible that no unexplained losses have occurred. Ideally, the ID should equal zero; but measurement errors will occur, and loss mechanisms that are difficult to measure do exist. (Such loss includes the surface contamination of process equipment or the small amount of trackout on employees' shoes.) These factors are considered when determining the total probable error and acceptable uncertainty from measurement errors and unrecognized loss mechanisms. If the ID exceeds these acceptable and probable limits, an unexplained loss has occurred.

The heart of nuclear inventory management is the preparation of a meaningful closed material balance. This task is difficult and involves many problems. Errors in management and interpretation of data must be understood, and the system itself must be designed to minimize these errors. An accurate system of records helps to minimize errors; thus, the Federal government requires that records be kept.

NUCLEAR MATERIAL RECORDS

The Code of Federal Regulations (10 CFR 70.51) requires that nuclear material inventory records be kept and specifies the information that the records must contain. With one exception, the form of the records is not specified; and, expectedly, the types and forms of records vary from facility to facility.

Usually, a simple but effective record system can be constructed from two types of records: (1) a basic record system that consists of a transfer journal, an inventory ledger, and a summary ledger and (2) an internal control record system that details and documents material movement and location within an MBA. Other special inventory records should be provided as needed.

THE BASIC RECORD SYSTEM

All material entering or leaving a facility must be accompanied by a "Nuclear Material Transfer Report" (DOE/NRC

Form 741). Since this form provides the record-keeping link for material movement between facilities, its format and required information are rigidly dictated by the Department of Energy (DOE) through use of the form and associated instructions contained in DOE Order 5630 (in draft) and a similar report by the Nuclear Regulatory Commission on nuclear material control. It is from this form that basic record system entries are extracted. Sections of the form must be filled in for the various licensing and identification data, for shipper's data, and for receiver's data. The form itself references 10 CFR 30, 40, 50, 70, and 150.

The purpose of the basic record system is to document receipts and removals of nuclear material for the entire facility.

The transfer journal (Figure 24-5-1) is the chief record of receipts and shipments. The essential items to be recorded are —

- Date of physical movement of the material.
- Transaction series and number.
- Weight of the nuclear material transferred.
- Percent of fissile isotope or isotopes in the material.
- Weight of the fissile isotope(s) transferred.

The transfer journal is designed around a division into a separate section for the shipper and a section for the receiver. Space for recording the shipper-receiver difference also is provided. Federal regulations require that the receiver of nuclear materials not accept the shipper's stated nuclear material content without verification. Any difference between the shipper's stated nuclear material content and the content as verified by the receiver must be recorded and reconciled if necessary. For nuclear safeguard purposes, the receiver must use his own measured value in subsequent accountability record keeping.

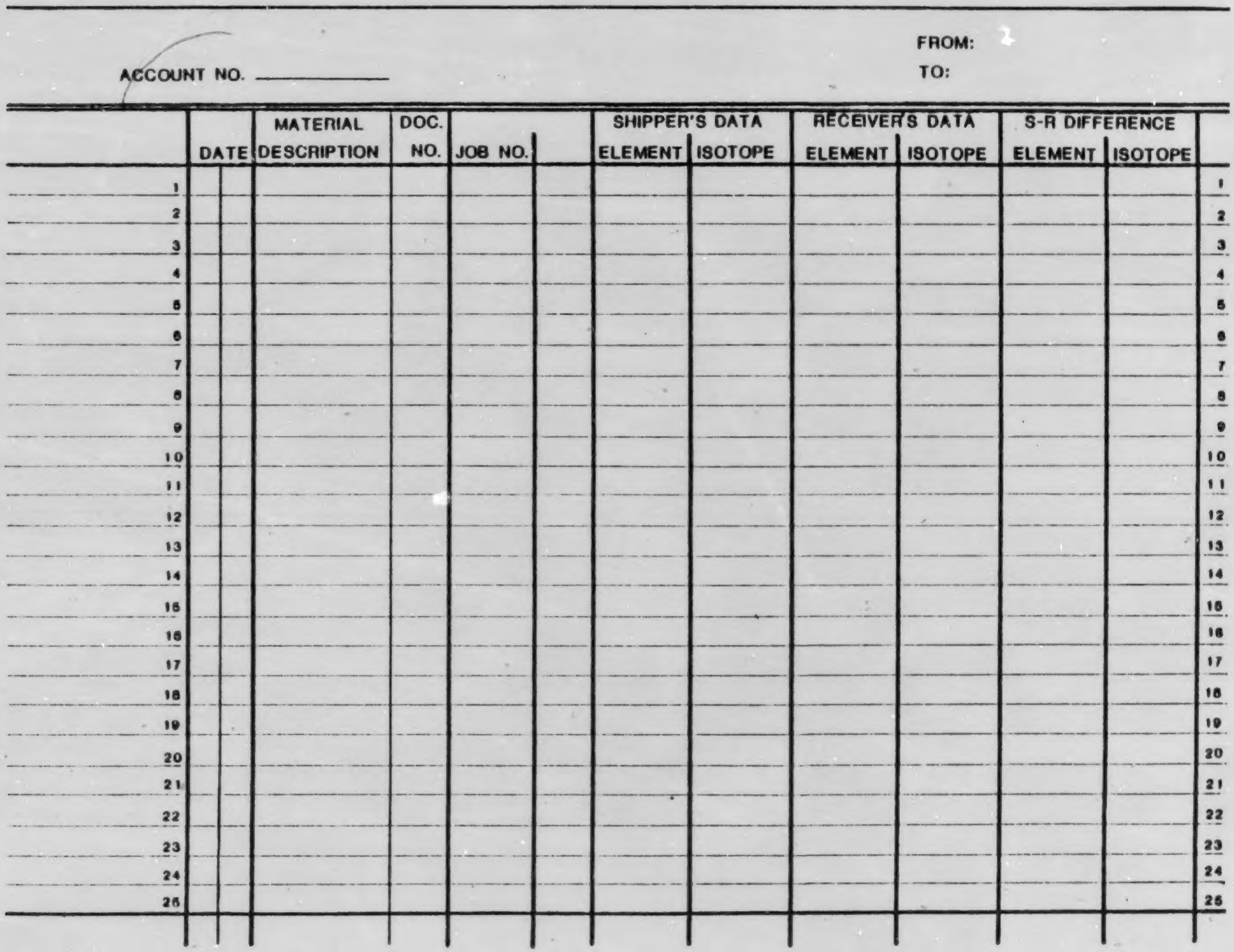
The inventory ledger provides a comparison to the transfer journal and is patterned after the standard accounting ledger. A typical page is illustrated in Figure 24-5-2.

The inventory ledger is a series of general ledgers. One general ledger is kept for each activity or contract; and, typically, one page is devoted to each accountable element and isotope. The page depicted provides space for identification information, including the contract number, the customer's account number, name, and address. Dollar values of the material may be included. Receipts, shipments/losses, and inventory are shown. This inventory is the inventory balance after that particular line of receipts, shipments, or losses has been accounted.

The depicted inventory ledger would require maintenance of a different page for each accountable element and isotope. The inventory column(s) of the inventory ledger sometimes is referred to as the "book" inventory for that element or isotope under the applicable contract. This value represents the amount of material for which the MBA respresentative is responsible under that contract.

A properly designed and maintained inventory ledger will provide the answer to most accountability questions. It will list how much material has been received for each contract, its dollar value, overall status of the project, losses, material shipped, dollar values of the material lost and shipped, and the ending inventory and its dollar value.

Addition of the summary ledger (Figure 24-5-3) makes the system a double-entry bookkeeping system. This helps to eliminate errors. The summary ledger is a summary material balance prepared from data transferred from the inventory



ACCOUNT NO. ____________

FROM:
TO:

	DATE	MATERIAL DESCRIPTION	DOC. NO.	JOB NO.	SHIPPER'S DATA		RECEIVER'S DATA		S-R DIFFERENCE		
					ELEMENT	ISOTOPE	ELEMENT	ISOTOPE	ELEMENT	ISOTOPE	
1											1
2											2
3											3
4											4
5											5
6											6
7											7
8											8
9											9
10											10
11											11
12											12
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15											15
16											16
17											17
18											18
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21											21
22											22
23											23
24											24
25											25

Figure 24-5-1. Typical transfer journal layout.

Contract No.
Customer:
Account No.

	Date	No.	Description	Receipts		Shipments/Losses		Inventory	
				Qty	$	Qty	$	Qty	$
1									
2									
3									
4									
5									
6									
7									
8									
9									
10									
11									
12									
13									
14									
15									

Figure 24-5-2. A typical inventory ledger form.

Beginning inventory	(from inventory ledger)
Receipts	
Total from ABC facility	(from transfer journal)
Total from DEF facility	(from transfer fournal)
etc.	
Total receipts	
Total to account for	
Removals	
Shipped to ABC facility	(from transfer journal)
Shipped to DEF facility	(from transfer journal)
etc.	
Total shipments	
Measured discards	
Other loss entries	
Total losses	
Ending inventory	(from inventory ledger)
Total accounted for	

Figure 24-5-3. Typical summary ledger.

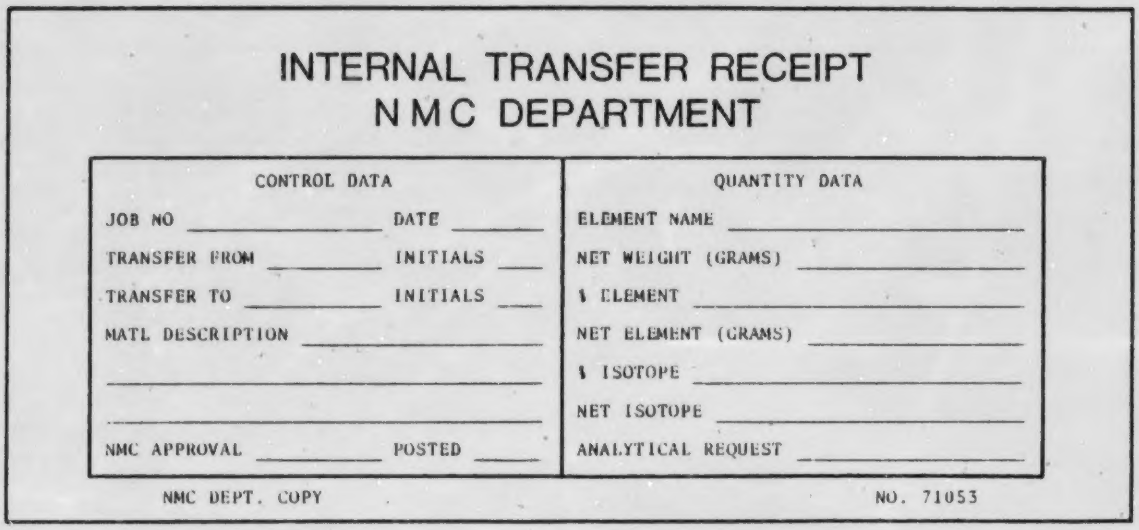
ledger and from the transfer journal.

Beginning and ending inventories for the current period are taken from the inventory ledger. Receipts, shipments, and losses are taken from the transfer journal. The total to account for must equal the total accounted for. If it does not, clerical errors and/or unrecognized losses have occurred.

Internal records are records internal to an MBA. Internal records are not as formal as records prepared for external use and scrutiny. Among them are internal transfer receipts, nuclear material discard or loss reports, monthly MBA reports, and inventory ledger.

The material status report (NRC/DOE-742) is filled out for material accountability and ending inventory. It is submitted at least by every March 31st and September 30th.

Internal transfer receipts document all movements of nuclear material between internal MBA's. There is no standard form, but the receipt should include a brief description of the material transferred (such as a batch number), any quantity data needed, and the authorizing signatures of the shipper and receiver. A typical form is shown in Figure 24-5-4.



INTERNAL TRANSFER RECEIPT
NMC DEPARTMENT

CONTROL DATA

JOB NO ______________ DATE ________
TRANSFER FROM __________ INITIALS ____
TRANSFER TO __________ INITIALS ____
MATL DESCRIPTION ______________
____
____
NMC APPROVAL __________ POSTED ______

QUANTITY DATA

ELEMENT NAME ______________
NET WEIGHT (GRAMS) ______________
% ELEMENT ______________
NET ELEMENT (GRAMS) ______________
% ISOTOPE ______________
NET ISOTOPE ______________
ANALYTICAL REQUEST ______________

NMC DEPT. COPY

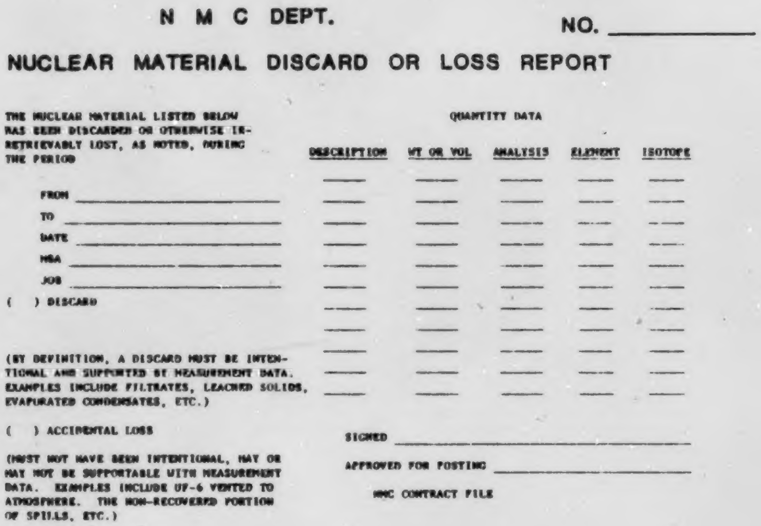
NO. 71053

Figure 24-5-4. Typical general-purpose internal transfer receipt.

Each MBA also must measure and report all losses and discards. A typical format is illustrated in Figure 24-5-5.

The inventory ledger maintained by the MBA shows receipts, shipments, discards or losses, and the resulting book inventory. MBA's also may keep transfer journals.

The monthly report (Figure 24-5-6) is made to the NMC



N M C DEPT. **NO. ___________**

NUCLEAR MATERIAL DISCARD OR LOSS REPORT

THE NUCLEAR MATERIAL LISTED BELOW HAS BEEN DISCARDED OR OTHERWISE IRRETRIEVABLY LOST, AS NOTED, DURING THE PERIOD

FROM ______________________
TO ______________________
DATE ______________________
MBA ______________________
JOB ______________________

( ) DISCARD

(BY DEFINITION, A DISCARD MUST BE INTENTIONAL AND SUPPORTED BY MEASUREMENT DATA. EXAMPLES INCLUDE FILTRATES, LEACHED SOLIDS, EVAPORATED CONDENSATES, ETC.)

( ) ACCIDENTAL LOSS

(MUST NOT HAVE BEEN INTENTIONAL, MAY OR MAY NOT BE SUPPORTABLE WITH MEASUREMENT DATA. EXAMPLES INCLUDE UF-6 VENTED TO ATMOSPHERE. THE NON-RECOVERED PORTION OF SPILLS, ETC.)

QUANTITY DATA

DESCRIPTION	WT OR VOL	ANALYSIS	ELEMENT	ISOTOPE
____	____	____	____	____
____	____	____	____	____
____	____	____	____	____
____	____	____	____	____
____	____	____	____	____
____	____	____	____	____
____	____	____	____	____
____	____	____	____	____
____	____	____	____	____
____	____	____	____	____
____	____	____	____	____

SIGNED ______________________
APPROVED FOR POSTING ______________________
NMC CONTRACT FILE

Figure 24-5-5. Typical nuclear material discard report.

Material Balance Area ________
Monthly Report for ________ 19 _____
(name of element)

	Element	Isotope
Beginning inventory	______	______
Receipts:		
from area ______		
from area ______	______	______
Total receipts	______	______
Total to account for	══════	══════
Removals		
to area ______		
to area ______		
discards	______	______
Total removals	______	______
Ending inventory	______	______
Total accounted for	══════	══════

Figure 24-5-6. Monthly MBA report format.

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department and summarizes receipts, shipments, discards, and inventory for the month.

PHYSICAL INVENTORY

Periodic physical inventories are necessary for effective inventory management. Without an inventory, the amount of material on hand or missing cannot be known. Information that a closed material balance can provide about unrecognized losses is made available only when the physical inventory has been taken and the results tabulated, totaled, and verified. Physical inventories are necessary to close the material balance.

In many facilities, complete physical inventories are scheduled semiannually for nonstrategic SNM and every two months for strategic SNM. More frequent inventories usually are not cost effective. The responsibility of taking the inventory may rest with the MBA custodians, special inventory management committees, or a group from the NMC office. Ideally, basic responsibility for the inventory should rest with the NMC group.

The first step in the preparation for a physical inventory is a review of the material likely to be on hand at inventory time. The review is performed to identify and correct anticipated problems. Inventory procedures then are written, and inventory teams are selected and trained.

In general, two types of inventories are possible — the "moving" inventory and the "shutdown" inventory. Production does not shut down completely during the moving inventory. Pauses for cleanup are made at individual stations or units, and each unit is inventoried before the next is begun. This type of inventory is not recommended since it is not likely to be as accurate as the "shutdown" inventory. The shutdown inventory demands that all processes within an MBA be stopped and cleaned up for physical inventory.

Some inventories are observed by internal or external auditors. These auditors observe the inventory procedures and examine and certify the tabulated data.

The data is listed on inventory data sheets, which may be processed electronically and combined with inventory stickers attached to each inventory item. As each item is inventoried, a sticker is placed upon it. This practice allows the checkers to verify that all items have been inventoried.

After inventory data is collected, it is audited to reduce the number of errors fed to the computer. Then a computer audit is performed to sort the data and to compare it with ledger values. Data is summarized, and an attempt is made to close the material balance.

MEASUREMENT METHODS

An inventory of each line item of nuclear materials requires special measurement techniques and procedures. These procedures involve three basic steps: (1) Total bulk quantity is determined by weighing or measuring volume of the material in a container; (2) the material is sampled; and, finally, (3) the sample is analyzed. Most solid materials are weighed. Liquids also may be weighed, although volume measurements are used as a more convenient method of bulk quantity determination. Sensitive scales are employed, and calibration-and-check weighing programs are established to ensure that the system gives accurate and reliable weights.

Newly developed force-measurement systems are accurate to better than 0.01%.

Liquid bulk quantity is determined by volume measurement. The volume of a container may be measured by sight gage, electromechanical float, manometer, ultrasonic level detector, differential-pressure (dp) cell, buoyancy probe, or with other devices.

After volume or weight of the material is measured, a representative sample must be taken. The first step is to mix the material in order that the sample will be truly representative. If the material is not mixed, the sample will yield inaccurate results.

Solids can be mixed in specially shaped blenders by a repetitive tumbling motion, by a specially designed motor-driven stirrer, or by pneumatic recirculation.

Liquids can be mixed by any of several methods: by motor-driven stirrer, by pumps used to recirculate and mix the liquid, or by "sparging" (in which air is pumped through the liquid to induce mixing).

After the material is mixed properly, any portion may be taken as a representative sample. Some sampling procedures require that 1/3 of the sample be taken from the top, bottom, and middle of the material. If the material is truly well-mixed, this procedure is unnecessary.

A question arises when physical inventory of a material in some container is based upon a single sample: How good (or representative) is that sample of the material? This question may be answered by comparing analytical results of the sample with analytical results of a second sample of the same material. The second sample should be acquired after additional mixing (and sampler line purging, where applicable) to ensure that the replicate sample was not obtained from the localized portion of the material out of which the first sample was drawn.

When representative samples of a material have been obtained, they are analyzed to determine the concentrations and compositions of the elements and isotopes contained.

Chemical analysis results then are applied to the bulk quantity value to determine physical inventory of the elements and isotopes in the material. Elemental inventories can be computed from Equation 24-5-2.

$$E = C \cdot Q \qquad \text{Eq. 24-5-2}$$

where:

E = the inventory of an elemental species of concentration C.

Q = bulk quantity of material being assayed.

For an accountable isotope whose element has an inventory of E, the isotopic inventory can be calculated from Equation 24-5-3.

$$I = FE = FCQ \qquad \text{Eq. 24-5-3}$$

where:

I = inventory of the isotope of weight fraction F (weight fraction = weight % ÷ 100).

If the isotopic composition is reported in units of mole fraction (mole % ÷ 100) instead of weight fraction, the weight fraction of the accountable isotopes must be computed (prior to inventory calculations) from Equation 24-5-4.

$$F = \frac{mM}{\text{average element weight}} \qquad \text{Eq. 24-5-4}$$

where:

m = mole fraction of the isotope of weight M and the average element weight (if not provided by the chemical analysis lab) is the sum of the product mM for all isotopes of the element.

Inventory quantities for all accountable elements and isotopes then are recorded on the physical inventory data sheet.

STATISTICAL EVALUATION

Physical inventories provide only an estimate of the material on hand; hence, a number of statistical techniques are used to evaluate measurement accuracy. Perhaps the most widely applicable of these techniques is the propagation of error analysis. This technique enables the technician to determine uncertainty of an inventory (or material balance) by providing a valid method for combining all uncertainties of the involved measurements. These measurement uncertainties are associated with instrument readings, chemical analysis determinations, and volume calibrations, among others. Such uncertainties are referred to as the standard deviation (S) of the measurement around the mean measurement value. The standard deviation of an inventory (or material balance),

$S_{\Sigma t}$, can be calculated from standard deviations of the individual inventory measurements (Equation 24-5-5).

$$S_{\Sigma i} = \sqrt{\sum^{k} S_i^2} \qquad \text{Eq. 24-5-5}$$

where:

k = individual inventory measurements.

i = 1.

Equation 24-5-5 is valid where the propagated error is describing only added (or subtracted) quantities. Each entry of S_i summed in Equation 24-5-5 is the result of propagated errors from the basic measurements used in calculating each inventory measurement. When basic measurements are multiplied, as in Equation 24-5-2, the propagated error of that elemental inventory measurement, S_C, could be calculated from Equation 24-5-6.

$$S_{CQ} = \sqrt{Q^2 S_C^2 + C^2 S_Q} \qquad \text{Eq. 24-5-6}$$

where:

S_C = standard deviation of the elemental concentration.

S_Q = standard deviation of the bulk quantity measurement.

When three basic measurements are multiplied as in Equation 24-5-3, the propagated error of the isotopic inventory, S_{FCQ}, can be calculated by Equation 24-5-7.

$$S_{FCQ} = \sqrt{C^2Q^2S_F^2 + F^2Q^2S_C^2 + F^2C^2S_Q^2} \quad \text{Eq. 24-5-7}$$

where:

S_F = the standard deviation of the isotopic weight fraction.

When two quantities are divided (as with specific gravity, SpG), correction of an indicated differential pressure liquid level, ILL, indication to a true reading is accomplished by use of Equation 24-5-8.

$$\text{True LL} = \frac{\text{ILL}}{\text{SpG}} \qquad \text{Eq. 24-5-8}$$

The propagated error for the True LL could be calculated from Equation 24-5-9.

$$S_{\frac{ILL}{SpG}} = \sqrt{\frac{(SpG)^2S_{ILL}^2 + (ILL)^2S_{SpG}^2}{(SpG)}} \qquad \text{Eq. 24-5-9}$$

where:

S_{ILL} = standard deviation of indicated liquid level.

S_{SpG} = standard deviation of specific gravity.

Despite the dissimilar appearance of the formulas, Equations 24-5-5, 24-5-6, 24-5-7, and 24-5-9 are derived from the same general equation. When combined uncertainty has been determined, it should be compared to the ID of the material balance. If ID exceeds combined uncertainty, it must be assumed that some part of the ID did not result from measurement error.

Other statistical methods are utilized to evaluate precision of single and duplicate measurements, precision of future measurements, probability that a stated percentage of future measurements will fall within a stated range, accuracy of an analytical procedure, and accuracy comparison of different measurement methods.

Statistical methods also are employed to evaluate sequential physical inventories and to detect any inventory trends; for example, a very slow loss (leak) or diversion. These methods include control charts, cumulative sum charts, and state estimation techniques (e.g., the Kalman Filter), among others. The student need be aware only that these techniques exist; their diversity, stage of development, complexity, and lack of universal application render them beyond the scope of this curriculum.

EVALUATION OF ID (INVENTORY DIFFERENCE)

The chief objective of the closed material balance system is determination and interpretation of ID. ID includes measurement uncertainties and unrecognized losses. An ID that exceeds total probable and acceptable uncertainty from measurement errors and unrecognized loss mechanisms is an abnormal condition and must be interpreted. Three steps are followed to evaluate ID:

1. Preliminary technical evaluation.
2. Statistical evaluation.
3. Detailed technical investigation of the problem areas involved.

Preliminary technical evaluation should determine whether or not a valid physical inventory was taken. Inventory data sheets should be scanned and any abnormalities noted. The technician should look for signs that quantities were estimated instead of directly measured, and the inventory should be checked for completeness. IDs from an MBA should be compared carefully to historical records for that MBA and a determination made whether the ID falls within the general range expected.

The second step, statistical evaluation of the observed ID should confirm and quantify conclusions of the preliminary technical evaluation. The propagation-of-errors method or control-chart techniques may be used. Observed ID values over a period of time may be charted to determine how they compare with current ID.

If the problem areas are uncovered during the preliminary technical evaluation and statistical evaluation, these areas must be investigated. If the ID did not pass the statistical test in the second step, or if it was determined to be of doubtful validity, it must be evaluated. It must be determined whether or not an unrecognized loss actually did occur. The following areas should be investigated:

- Completeness of physical inventory.
- Accuracy of quantity data.
- Accuracy of beginning inventory.
- Completeness of all pertinent records.
- Accuracy of entries on the records (especially of transfer data.)
- Possibility of localizing the loss.
- Accuracy of measurement uncertainties.
- Validity of inventory procedures.

This list is incomplete. Each situation is unique and often seems to require the combined skills of a detective, a statistician, and an accountant.

EXERCISES

1. A vessel contains 50 liters (±0.5 liters) of solution. The chemical analysis lab reports the following chemical and isotopic analyses in duplicate samples:

	Sample 1	Sample 2	Uncertainty
U	102.3g/1	102.7g/1	±0.4 of reported value
^{233}U	0.15%	0.15%	±0.005%
^{234}U	0.07%	0.08%	"
^{235}U	21.31%	21.29%	"
^{236}U	0.015%	0.020%	"
^{238}U	78.45%	78.46%	"

   Compute the average analysis, and inventory the vessel for the accountable element and isotopes.

2. Perform a propagation-of-error analysis on the inventroy of material in the vessel described in Exercise problem 1.

MATERIALS

Mass balance.
Vessel containing solids.
Calculator.
An instrumented, calibrated metal tank containing liquid of specific gravity (SpG) 1.1 to 1.2. Instrumentation should be ΔP cell with appropriate ΔP transmitters (2) and level and SpG recorder. (Recorder output should be precalibrated for tank volume.)
1-liter graduated cylinders (3 each).
0.8 liter of solution.

PROCEDURES

PHYSICAL INVENTORY DETERMINATION

1. Weigh a filled container "S" (of known tare weight), and calculate net weight.
2. Apply a provided elemental and isotopic analysis, to determine "inventory" of that container for the accountable SNM: U, ^{233}U, and ^{235}U.
3. Record data in Data Table 24-5-1. Determine limit of error on inventory in "S."
4. Read liquid level and SpG off recorder.
5. Apply SpG correction to measured liquid level.

$$\text{True Liquid Level} = \frac{\text{Measured Liquid Level}}{\text{SpG}}$$

6. Apply given true liquid level reading to calibration curve or chart to determine volume in tank.
7. Apply tank volume to analytical data to determine assay of accountable elements and isotopes in tank.
8. Record on physical inventory table (Data Table 24-5-1).
9. Determine limit of error on inventory in tank.

VOLUME/MATERIAL BALANCE AROUND A TRANSFER

Use three graduated cylinders, A, B, and C. A should be near full of solution "containing" SNM; B and C should be empty.

1. Inventory accountable material in A, B, and C.
2. Pour about half of solution from A into B; empty B into C.
3. Inventory accountable material in A, B, and C again.
4. Compare total measurable volume (A + C, B should contain only unmeasurable "process hold-up") with initial volume in A before transfer.
5. Compare total inventory of accountable elements and isotopes before transfer with those after transfer.
6. Determine if pretransfer and post-transfer volumes and inventory values agree within the limit of propagated error.
7. Record post-transfer inventory on physical inventory table (Data Table 24-5-1).
8. Compare total physical inventory with ledger inventory. Determine if difference between the two is greater than propagated limit of error for all the physical inventory measurements. If physical inventory does not agree with ledger values, itemize actions that

should be taken to correct the problem and/or to report the condition.

DATA TABLE

DATA TABLE 24-5-1
PHYSICAL INVENTORY (IN GRAMS)

Vessel	Location	U	^{233}U	^{235}U	Pu	^{239}P	^{241}P
Total							
Ledger Inv.							
Inv. Difference							
Limit of Error							

REFERENCE MATERIALS

Lovett, James E. Nuclear Materials Accountability Management Safeguards. Hinsdale, IL: American Nuclear Society, 1974.

Proceedings of the Annual Meeting of the Institute of Nuclear Materials Management, Inc. Louisville, KY: INMM Publications Office.

U.S. Nuclear Regulatory Commission. Rules and Regulations; Title 10, Code of Federal Regulations, Parts 0 to 199. Washington, DC; Superintendent of Documents, U.S. Government Printing Office, January 1980, or current issue with latest Supplements.

SECTION I – MATCHING

Match each term in column A with the proper definition or description in column B.

Column A

_____ Nuclear material inventory management.

_____ Source nuclear material.

_____ Special nuclear material.

_____ Strategic special nuclear material.

_____ Material balance area.

_____ Closed material balance.

_____ Transfer journal.

_____ Summary ledger.

_____ Inventory ledger.

_____ Internal records.

_____ Moving inventory.

_____ Shutdown inventory.

Column B

A. Performance of nuclear material inventory measurements, maintenance of verifiable records, provision of reports, performance of data analysis to account for nuclear material.

B. Includes Pu, ^{233}U, and the element uranium enriched in the isotope ^{233}U or the isotope ^{235}U.

C. Includes the element uranium enriched to 20% or more in the isotopes ^{233}U or ^{235}U.

D. An identifiable physical area through which measured quantities of SNM are moved.

E. Includes depleted thorium or natural uranium.

F. The system used to establish optimum accountability for SNM.

G. The chief record of receipts or shipments.

H. A series of general ledgers.

I. Prepared by taking data from inventory ledger and transfer journal.

J. Records internal to an MBA.

K. Demands that all processes within an MBA be stopped.

L. Allows production to continue.

SECTION II

1. State the equation for the closed material balance.

2. Describe material unaccounted for; cite its significance to the closed material balance.

3. Describe the basic record system.

4. List the records internal to a material balance area.

5. Describe the purpose of and necessity for the physical inventory.

6. List and describe two physical inventory techniques and the advantages and/or disadvantages of each.

7. What three steps are basic to the measurement of SNM?

8. What three steps should be followed to evaluate ID?

SECTION III – STATISTICAL EVALUATION

The MBA "Lab" is responsible for 1500 grams of Pu, and 1320 grams of ^{239}Pu based on its ledger value. All of the Lab's accountable material is of the same isotopic composition measured as 87.95 wt % (±0.01 wt %) ^{239}Pu, and no other accountable isotopes are present in measurable quantities. All containers bearing SNM are tabulated below:

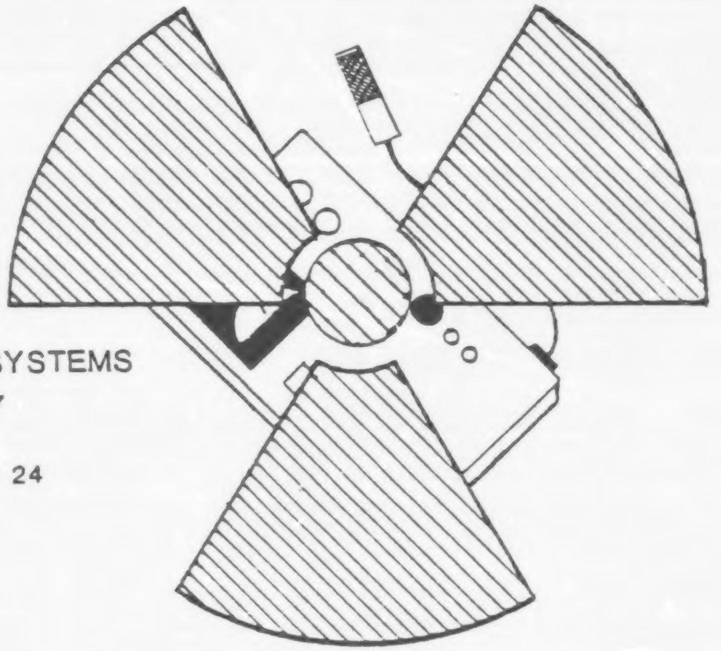
Container ID	Net mass (g)	Pu Concentr (g Pu/g solid)
A	857.2 (±0.05g)	0.883 (±0.4% of reported value)
B	312.1 "	0.881 "
C	602.5 "	0.779 "

1. Tabulate physical inventory.
2. Perform error propagation to determine if the Lab has all material for which it is responsible.

NUCLEAR TECHNOLOGY

MODULE 24 - 6

NUCLEAR PLANT SAFEGUARDS



NUCLEAR SYSTEMS
AND SAFETY

COURSE 24

CORD CENTER FOR OCCUPATIONAL RESEARCH AND DEVELOPMENT

COURSE MODULES

COURSE 24 NUCLEAR SYSTEMS AND SAFETY

MODULE 24-1	Neutron Physics
MODULE 24-2	Cross Sections and Reactions
MODULE 24-3	Fission Rate and Reactor Power
MODULE 24-4	Nuclear Criticality (0530 ERDA Manual)
MODULE 24-5	Nuclear Material Inventory Management Techniques
⇨ MODULE 24-6	Nuclear Plant Safeguards

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INTRODUCTION

Nuclear plant "safeguards" is a term that describes a technology that recently has evolved into an important and necessary segment of the nuclear industry. Safeguards are those devices and contingency plans that provide for the following:

- Physical protection of nuclear material at fixed sites and in transit to and from plants in which the nuclear material is used.
- Protection of nuclear material against theft.
- Protection against acts of industrial sabotage.

The term "sabotage" implies a willful effort by a civilian or by an enemy agent to destroy. Sabotage has been the intent of many contemporary terrorist groups. An act of sabotage involving a nuclear power plant could release quantities of radionuclides many times over the maximum permissible concentrations currently deemed safe.

Theft prevention is also a prime responsibility of effective safeguards, assuming the threat that nuclear weapons possibly could be made by terrorists if the required nuclear materials are available. Much information, non-nuclear materials, and equipment required for the designing and building of a variety of types of fission explosives are readily available. With appropriate (and obtainable) technical skills and resources, it is conceivable that a few persons, who possessed about 10 kilograms of plutonium or uranium-233 oxide, or two dozen kilograms of highly enriched uranium oxide and a substantial amount of high explosive, within several weeks could safely design and build a crude, transportable fission bomb – one that possibly could explode with a yield equivalent to at least 100 tons of high explosives and that could be carried in an automobile. Any type of plutonium, highly enriched uranium, or uranium-233 now used or contemplated for use by the nuclear industry might serve this purpose.

Because of the toxicity of fissionable materials in the environment and the possibility of a terrorist group constructing a nuclear weapon or committing sabotage, nuclear plant safeguards, by necessity, must employ the most stringent and analytical techniques available to anticipate and deter any threat to a nuclear facility.

MODULE PREREQUISITES

The student should have completed Course 1, "Radiation Physics"; Course 2, "Radiation Protection I"; Course 11, "Radiation Detection and Measurement"; Course 17, "Radiation Protection II"; and Course 23, "Nuclear Chemical Processes." The following courses will prove helpful: Course 3, "Principles of Process Instrumentation," and the later Course 25, "Radioactive Material Handling Techniques."

OBJECTIVES

Upon completion of this module, the student should be able to:

1. Define the following terms:
    - Nuclear plant safeguards.
    - SNM.
    - Physical barriers.
    - Protected areas.

Vital equipment.
Isolation zone.
Material access area.
Safeguard contingency plan.
SSNM.
Authorized person.
Repeated threats.
Diversion.
Single threat.
Proliferation.
Denial mode operation.
Accountability.
Precision.
Accuracy.
MBA.
Physical inventory.
Book inventory.
MUF.
Confidence level.
LEMUF.
"EASI."

2. List and describe general characteristics of the materials that must be safeguarded.
3. List and describe, in general terms, federal requirements concerning nuclear safeguards and where they can be found.
4. List safeguards goals, threats, responses to threats, methods of prevention, and constraints on safeguards systems.
5. Describe how to accomplish physical security for the site, the control of personnel, the control of material, and overall safeguards control.

6. Calculate book inventory, MUF, LEMUF, and uncertainty on sums and differences (error propagation).
7. Determine effectiveness of a certain safeguard system; identify developmental steps in the establishment of an effective system.

GOVERNMENTAL REGULATIONS

Radioactive materials required by law to be safeguarded are called "special nuclear materials (SNM)" and include:

- ^{235}U (contained in uranium and enriched to 20 percent or more in the ^{235}U isotope).
- ^{233}U.
- Plutonium, by itself or in any combination in a quantity of 5,000 grams or more.

Title 10, Part 73, of the Code of Federal Regulations (10 CFR 73) describes requirements for the physical protection of SNM in transport (whether by land, air, or sea), production, and utilization. This module is concerned with safeguards involved in plant site facilities (production and utilization).

Basically, the regulations concerning physical security encompass seven general areas:

1. The physical security organization.
2. Physical barriers.
3. Access requirements.
4. Detection aids.
5. Communication requirements.
6. Testing and maintenance.
7. Response requirements.

THE PHYSICAL SECURITY ORGANIZATION

An organized method of site protection is required. This organization is achieved by the use of guards and watchmen whose primary responsibility is protection of SNM against theft and protection of the plant against sabotage. Members of the security organization must be trained, equipped, and qualified for their specific jobs. Specific criteria for job performance are found in Appendix B, 10 CFR 73.

PHYSICAL BARRIERS

Physical barriers refer to fences constructed of No. 11 American wire gage, topped with barbed wire; building walls constructed of stone, concrete, steel, or comparable materials; and ceilings and floors of building constructed in such a manner as to offer substantial resistance to entry by unauthorized personnel. Physical barriers usually surround locations into which access is controlled or regulated; these locations are known as "protected areas."

Vital equipment – any equipment, system, device, or material that could endanger public health and safety if it were damaged or destroyed – is required to be located within a protected area. Access into the area requires passage through at least two physical barriers. An isolation zone must surround the protected area and be monitored or checked periodically to detect presence of unauthorized or suspicious persons or equipment. The isolation area surrounds the perimeter of the protected area and is clear of all objects that could conceal or shield an individual. It must be well lighted.

The term "material access area" describes any location that contains SNM. It also is protected by physical barriers.

Figure 24-6-1 depicts the different areas involved in the physical protection of a site.

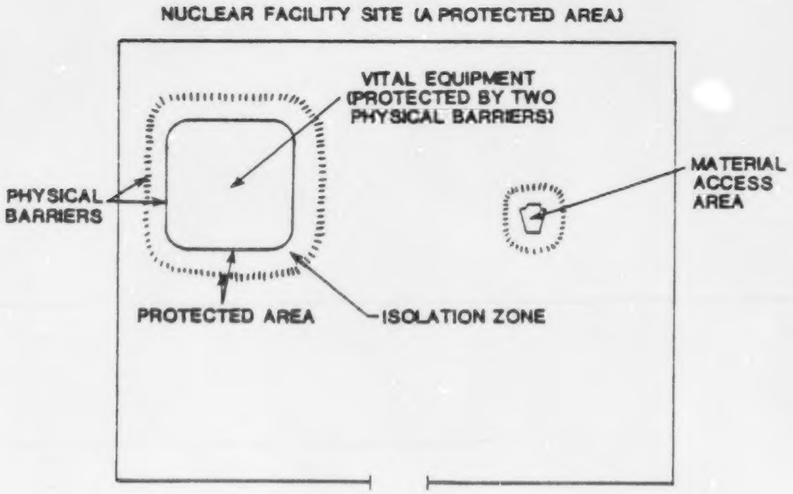
NUCLEAR FACILITY SITE (A PROTECTED AREA)
VITAL EQUIPMENT (PROTECTED BY TWO PHYSICAL BARRIERS)
PHYSICAL BARRIERS
MATERIAL ACCESS AREA
PROTECTED AREA
ISOLATION ZONE

Figure 24-6-1. Areas involved in site protection.

ACCESS REQUIREMENTS

"Access" refers to all points through which personnel and vehicles, including shipping and receiving, must pass. At these points all individuals, except employees who possess personnel security clearance from the Nuclear Regulatory Commission (NRC) or the Department of Energy (DOE), are searched for firearms, explosives, etc., which possibly could be used for sabotage. Non-employees are escorted at all times while in a protected area.

Packages delivered into protected areas are checked for proper identification and authorization and are searched at random intervals.

A badge identification system is utilized for all personnel. Those individuals authorized to enter vital areas and material access areas are issued specially coded badges.

All keys, locks, and combinations are closely controlled and changed whenever there is any evidence of tampering. When an employee terminates employment, all keys, locks, and combinations to which that employee had access are changed.

DETECTION AIDS

Alarms are used to alert security personnel to any unauthorized intrusion into protected areas, buildings, vital area, or materials access area. The alarms are required to annunciate in a continuously monitored control alarm station. Performance and reliability levels of the alarms are specified in 10 CFR 73.

COMMUNICATION REQUIREMENTS

All guards and watch personnel are required to have continuous two-way communication with the central alarm station. Two-way radio capability in addition to conventional telephone is required between the facility and local law enforcement authorities. All communication equipment should have an independent power source in the event the primary power supply becomes inoperative.

TESTING AND MAINTENANCE

All communications equipment is to be tested at the beginning of each security personnel shift. Alarms, physical barriers, and other security devices are tested at least every seven days.

RESPONSE REQUIREMENTS

A nuclear facility is required to file a safeguard contingency plan for dealing with threats, thefts, and industrial sabotage. The contingency plan identifies perceived dangers and incidents with which the plant will deal and the general methods for dealing with the dangers. Along with criteria used for the initiation and termination of responses to safeguard systems, the following descriptions also are required to ensure adequate security:

- Physical layout of the plant, describing location of structures on the site and transportation routes available to the structures.
- Hardware (such as alarms, locks, seals, area access, armaments, communication and surveillance devices, etc.) in the security system.
- A listing of local law enforcement agencies and their response capabilities.
- A responsibility matrix that lists agencies or persons responsible for each decision and action associated with specific responses to safeguard contingencies.

SAFEGUARD GOALS AND RESPONSES

Safeguard procedures for SNM are necessary because of the attractiveness of nuclear material for explosive devices and sabotage. This attractiveness is maximized in regard to high fissionable (fissile) material such as ^{235}U and ^{239}Pu because only a small amount need be stolen to assemble a nuclear weapon. These materials are known as strategic special nuclear materials (SSNM). Uranium enriched 20% or more and all forms of plutonium enter into this category.

The attractiveness of theft of fissile materials is inversely proportional to the effort required to convert the material to usable form. (PuO_2 mixed with UO_2 is not as attractive as pure PuO_2 because the chemical separation of the plutonium from uranium is required.)

Vulnerability of a site is defined by the different potential threats possible, such as diversion of SNM, sabotage, theft, etc. The "correct" response to a safeguard threat is evaluated according to the seriousness of the offense and the potential consequences. These two factors – vulnerability and response – therefore will delineate safeguard goals, responses, methods of prevention, and constraints in the safeguard system.

The major objective of nuclear safeguards is to provide protection for the site of the facility. The site has a fixed physical boundary that is secured by a double barrier. All entrance and exit points are monitored by security guards, who also conduct patrols around the site perimeter. The whole idea is to keep unauthorized people out; an authorized person, on the other hand, is one who has been designated in writing by the plant administration to have some responsibility with regard to SNM.

Areas of the site boundary that are not guarded by personnel should be monitored by at least two or three different types of electronic sensors and/or surveillance devices. This equipment includes sonic detectors (which sound an alarm if "triggered" by noise over a specified decibel level), TV cameras known as "CCTV" (closed-circuit TV on automatic scan), magnetic detectors, etc.

Guarded portions of the site should be monitored by at least two guards per entry (in bullet-proof enclosures). All entries and exits also should be equipped with radiation and magnetic monitors, which detect radioactive material leaving the site and the entrance of metal objects, such as guns or knives.

All transportation into and out of a facility must be authorized and continuously monitored and accompanied by full-time security personnel. All exits should lead through radiation monitors.

Because of the function of a nuclear facility and the potentially hazardous materials involved, certain objects should not be allowed into the site unless for the expressed and authorized use of site security. Such objects include cameras, firearms (or any type of potential weapon), explosives, and toxic materials or chemicals.

Strategically located sensors and surveillance devices and randomly timed security patrols inside the facility, in addition to random, though frequent, tests of the safeguard systems, are important characteristics of any safeguard system. The randomness of the tests and patrols creates a significant unknown variable to potential intruders. The tests themselves must be closely monitored and continuously evaluated to ensure that methods, strategy, status, or limitations of the system are not revealed.

Specific safeguard goals vary for different threats. For sabotage of the facility or site and/or SNM diversion, the primary goals are –

- Stop the attempt before it begins.
- If stopping the attempt is not possible, delay the attempt until security reinforcements can be alerted.
- Prevent the threat from being carried to completion.

When theft of SNM is attempted, whether unsuccessfully or otherwise, pertinent safeguard goals are –

- Detect the unsuccessful attempts.
- Estimate possible loss of the attempted diversion or claimed theft; that is, assess the credibility of an alleged threat.

If a definite loss of SNM is suspected or if a successful attempt has been made, the primary goals are as follows:

- Estimate the quantity, form, and previous location.
- Retrieve it before it leaves the site.

In the establishment and implementation of goals, four considerations are important. The safeguards should –

1. Be independent of the particular type of nuclear facility in which they are implemented. All possible contingencies should be included in every safeguard strategy and not be simply a function of a particular portion of the nuclear fuel cycle; for instance, reprocessing or waste disposal.
2. Be subtle and discreet so as not to be flagrantly observable to the nuclear operators. Such discretion prevents interference and avoids schedule shutdowns.

3. Minimize disruptions of the normal day-to-day operations of the facility in order that disruption cannot be used to the advantage of an intruder.
4. Be economically feasible.

THREATS AND SAFEGUARDS

There are two basic types of threats: repeated and single.

Repeated threats are small and covert. They require special knowledge of the facility and its operations. These threats usually are instigated and attempted by authorized personnel with or without the aid of unauthorized personnel. Since the theft or diversion is repeated, personnel involved will wish to escape detection and not be caught. (A "diversion," for purposes of this module, is defined as "unauthorized operations and movement of material within authorized areas." Diversion becomes theft when the diverted material leaves the authorized area, not necessarily the site.)

A large "single threat" is defined as being overt. A threat of this nature implies armed attack, a take-over of the facility, and/or large-scale theft.

An armed attack is likely to be initiated by a group of terrorists dedicated to sabotage of the facility and/or theft of SNM.

Take-over of a facility is likely to be in a country nationalizing an internationally run facility. This type of situation does not now exist but has been hypothesized. As in the case with many perceived threats, scenarios are theorized to formulate appropriate responses and outcomes.

A large-scale theft involves enough nuclear material for at least one clandestine weapon. Although some knowledge of the facility and its operations would be required in a large-scale theft, inside aid would not be deemed essential.

Vulnerability of a facility to a specific threat is largely dependent upon the nature of the threat itself. For instance, a nationalized facility probably will be completely vulnerable to take-over due to the magnitude of the conspiracy and the bureaucratic level at which the threat takes place. A take-over, due to nationalization, is likely to be a matter of non-weapon states (countries) attempting quickly to acquire weapons.

The definition of sabotage implies probable facility damage. Theft of SNM may or may not present property damage, depending upon the manner in which the theft is organized and instituted. It is evident that actual physical vulnerability is a function of the threat against the site and the techniques used to accomplish that threat. Therefore, the proper response of a safeguard system to a threat is dependent also upon the seriousness of the crime and the potential consequences. A response may range from a "quiet, inhouse" investigation to a nationwide search. Authorities who manage the response then will vary from the employee's supervisor, plant management, plant security force, to law enforcement agency.

A high level of nuclear weapons production or the spread of nuclear weapons among previously nonproducing states is known as "proliferation." As mentioned earlier, because the "current state of the art" of assembling a nuclear weapon is so widely disseminated, the greatest preventive measure available to nuclear facilities is the physical treatment of fissionable materials. Such treatment includes –

- Denaturing of fuel by adding ^{238}U to ^{233}U, which, in effect, dilutes the isotope.
- "Spiking" of the fuel by adding highly radioactive materials such as ^{60}Co to the fuel materials, making it "too hot to handle."
- Coprocessing of the plutonium and uranium, which does not allow amounts of plutonium to become concentrated in one location.

Technically advanced safeguard systems, such as those discussed below, are utilized in regard to the handling of fissile materials:

- Fuel cycle centers in which spent fuel or ore is enclosed in a single location and processed into the final fuel.
- Process modifications in which all pure fissionable process operations are located behind 4-ft concrete walls.
- Use of computers to provide inventory data within 15 seconds.
- Coordinated command, control, and communications center (C^3) through which all process operations are monitored offsite.
- A continuous inspection program that monitors, by the use of a full-time inspector, all plant operations, especially those that involve SNM movements.
- Denial mode operation involving a C^3 system that allows for removal or control of all, or a portion of, process equipment from operators and limits access to parts of, or all of, the facility.

The last preventive measure listed above – denial mode operation – also is known as "active use-denial." Active use-denial has a mechanized counterpart known as "passive use-denial." Passive use-denial involves use of computers that automatically "scram" a system if more than "x" amount of fissionable material is unaccounted for or if an operator either does not give a properly coded password or otherwise exceeds the limits of his or her authority.

Nuclear plant safeguards are limited by certain restrictions, most of which have been discussed earlier in this module. These constraints are, in reality, guidelines by which goals are to be designed and implemented. They involve these factors:

- Economic feasibility of a safeguards system.
- Maintenance of the "normal" plant operations and capabilities.
- Use of the most current safeguard technology available.
- Utilization of active/passive use-denial.
- Human evaluation of system designs, functions, and reliability.
- Extension of responsibility of SNM in order that no one person maintains complete control.
- Inclusion of safeguards into the earliest design stages of plant operations.
- Continual SNM accounting (in terms of quantity and location).
- Minimization of operational disruptions.
- Denial of access to SNM.
- Independence of safeguard facilities in relation to the fuel cycle.

- Ability of safeguard systems to function efficiently without knowledge of the operators.

In light of the previous discussion on nuclear safeguards, five major conclusions should be drawn regarding the nature and scope of an effective safeguard system:

1. There is no 100% foolproof scheme of safeguards available.
2. The primary directive of a well-engineered safeguards system is to delay the adversary by stacking barriers (defense-in-depth).
3. The system must be demonstrated frequently without revealing status of strategy.
4. Capability must exist for detection of unsuccessful threats.
5. The system should be sensitive and expeditious in detecting even small losses of SNM.

PERSONNEL SAFEGUARDS

Security Personnel

A substantial portion of any safeguard system is its personnel. Although human involvement allows for immediate decision-making capability and physical control of a situation, it also is capable of participation in covert threats, erroneous judgment, and neglect of duty. It is, therefore, most important that security personnel be appropriately chosen and trained for their specific duty — the protection of a nuclear site.

Although there are currently no federal standards applicable to the training of nuclear operators, there is a specific criterion used concerning security forces. It is found in Appendix B, 10 CFR 73.

Before an individual can become an employee of a nuclear security force, he/she must receive a Department of Energy (DOE) or Nuclear Regulatory Commission (NRC) clearance, have no felony arrests, no "violent" political ties (to a radical organization), and be trustworthy. (Because of the subjective nature of the human species, it is understandable that some of the criteria used for evaluating individuals is somewhat subjective also.)

Training of security forces involves several different subject areas. Because of plant automation, it is imperative that security personnel (SP) be trained to interact with computers. In addition, there is training concerning local security rules and regulations and the basic federal regulations involved with nuclear security. Operations training, including radioactivity, equipment operations, safety, and specific security systems utilized at a site, complete the training of SP.

Personnel Control

Several methods are employed to ensure internal facility security. Basically, the objective in internal security is to limit personnel access to process and equipment — to limit employees to those areas in which they have a specific duty. This is achieved by direct administrative control and correct placement of physical barriers within a facility. Restriction of personnel to their own job areas limits opportunities for unauthorized manipulation of equipment and movement of material. Active surveillance and strategic placement of sensors decrease the probability of unauthorized movement of SNM.

Entrance into a nuclear facility is limited to authorized personnel who possess full security clearances. Unauthorized personnel must arrange with plant management for access into the plant at least 24 hours in advance. Individuals who are not United States citizens may be denied access for any reason. If admitted into a facility, they are required to have a continuous escort and to be under constant surveillance.

Identification badges usually are required for entry into plant facilities, as is positive identification of the entrant by one or more guards (not simply recognition, but positive identification). Other proofs also may be required, such as voice-print verification, fingerprints, signature, and weight. (Weight upon entrance is checked against weight upon exit to prevent shielded material from being carried out.)

Exit from a nuclear facility requires a radiation check, metal monitor check, badge exchange, identification, and a search of all personal effects including handbags, brief cases, and lunch pails. A random hands-on "pat down" search should be conducted, especially if a person acts suspiciously.

Internally, electronic records should be maintained as to who enters and exits and at what time.

Doors should be equipped with locks (as specified in 10 CFR 73) and the number and location of keys limited.

Any time that access is gained to sensitive materials or equipment, at least one authorized person should be present. If access is gained to SNM or SSNM, at least two authorized persons need to be present. The area surrounding SNM should be equipped with sensors, surveillance detectors, and seals to maintain constant vigilance.

ACCOUNTABILITY

Computers integrated into a safeguards system not only reduce the opportunities for SNM and SSNM diversion, but also improved safety factors because a quick response could prevent criticality. Automated computerized systems also reduce personnel exposure to radiation, improve nuclear process control, and reduce dependence upon personal subjective reliability and judgment. Another area in which computers have proved to be invaluable is accountability.

Accountability is the term that describes record-keeping procedures on amounts and locations of nuclear materials; quantitative determination of material transfers, receipts, shipments, and losses; and static inventories (those that do not change). Such record keeping is a government requirement for any facility that handles nuclear material.

Accountability is similar to one's reconciling a checkbook at the end of every month, except that in a nuclear facility the books must be "balanced" as often as possible. With computers, accountability is possible every few minutes.

A materials accounting system has several basic criteria. One deals with precision, defined by an amount "X" ± another amount "σ." Accuracy describes how close to the true or correct value the answer is. Unlike a personal checkbook, however, nuclear accounting is plagued with uncertainties (unknown or unascertainable differences) due to the fact that actual quantitative measurement often is difficult or extremely hazardous to obtain.

The very nature of the accounting reports requires continuous monitoring and verification of SNM process operations and SNM storage locations, along with monitoring and verification of SNM inventories.

Each accountability function is performed independently and separately in an area known as a "material balance area" (MBA). An MBA is a physically definable area inside or outside a facility in which the physical inventories (how much material is on hand at the present) and material movements into, and out of, the area are quantitatively determined. The MBA may be as small as a unit process (an evaporator) or as large as a small pilot plant (i.e., solvent extraction plus head-on dissolution and tail-end fabrication.)

When the physical inventory has been accepted by all authorities, including the government, it becomes the book inventory or the last inventory. The new inventory, taken after a specified period, then is called the "ending inventory." For a single time period, then, one has —

> the book or last inventory (adjusted
> for additions and removals) for the
> beginning of the time period
>
> +
>
> the physical or ending inventory (for
> the end of the time period).

The material balance then is defined as "the material unaccounted for" (MUF), or —

MUF = last inventory (LI) + additions
to the facility (A)

minus (-)

ending inventory (EI) + disbursements
or removals from the facility (D)

$$MUF = (LI + A) - (EI + D)$$

Equation 24-6-1
(Equation = Eq.)

To use the checkbook analogy, a bank balance = (previous balance + deposits) - (final balance + withdrawals).

Instead of dollars, SNM accountability is measured in kilograms (kg) of fissionable material.

EXAMPLE

Given: The following information:

Last Inventory	506.8 kg
Additions	29.4 kg
Ending Inventory	512.7 kg
Disbursements	8.1 kg

Find: Book inventory.

Solution: Uncertainty is a statistical measurement that calculates the possible degree of error in a given situation. It is expressed as ± "X amount" kg of material, X representing a quantitative value. For example: Last inventory = 506.8 kg ± 14.6 kg (X = 14.6). Uncertainty is represented by the Greek symbol sigma (σ), and the total uncertainty (σ_{Σ}) is defined as the square root of the sum of the uncertainty values squared. For example:

Last Inventory	506.8 kg ± 14.6 kg	(σ_1)
Additions	29.4 kg ± 2.0 kg	(σ_2)
Ending Inventory	512.7 kg ± 1.8 kg	(σ_3)
Disbursements	8.1 kg ± 0.3 kg	(σ_4)

$$\sigma_\Sigma = \sqrt{(\sigma_1)^2 + (\sigma_2)^2 + (\sigma_3)^2 + (\sigma_4)^2}$$

$$= \sqrt{(14.6 \text{ kg})^2 + (2.0 \text{ kg})^2 + (1.8 \text{ kg})^2 + (0.3 \text{ kg})^2}$$

$$= \sqrt{220.49 \text{ kg}^2}$$

$$\sigma_\Sigma = 14.84 \text{ kg.}$$

The uncertainty for the ± values is known as "error propagation." The book value equals 528.1 kg ± 14.84 kg.

Inventories normally are calculated to a 95% confidence level. This is a value by which one can be 95% certain that the inventory value obtained is correct. The 95% confidence level imposes a limit of error on materials unaccounted for (LEMUF) and may be defined as 2σ.

$$\text{LEMUF} = 2\sigma \qquad \text{Eq. 24-6-2}$$

If MUF > LEMUF, a diversion is likely to have occurred (95% certain).

The preceding information is synthesized in the following paragraphs. (The below Example is included in the Procedures section of this module.)

Calculation of Probability for Interruption of an Intruder

A statistical computer program has been developed to ascertain probability of interruption by an intruder (unauthorized personnel). This program, known as "EASI," is based upon two factors:

- Probability of certain events happening (a device detecting, a security force interrupting an intruder, etc.).
- Uncertainties of response times (for the intruder, alarm devices, and action by a security force).

When the final step-by-step input is complete, the final number obtained is the probability of interruption, P(I).

The "EASI" program, instructions on how to actually place the program into a calculator, and an example of a "Potential Intruder Sequence," are provided in Procedure 1, "Evaluation of Physical Security by Use of the EASI Computer Program." Terminology and abbreviations used in the program are defined below.

E(tr) – Estimated time of response (of security forces), in minutes.

σ(tr) – Uncertainty on the estimated response time, in minutes.

P(c) – Probability of appropriate alarm/security communications.

E(ti), E(t1),E(t2), ... etc.) – Estimated time for intruder(s) to complete a given operation, in minutes.

σ(ti),σ(t1),σ(t2), ... etc.) – Uncertainty on estimated time for intruder(s) to complete a given operation, in minutes.

P(d) [and P'(d) and P''(d)] – Probability for device(s) detecting an intruder. (This does not refer to the act of interrupting, which is done by Security.)

P(I) – Probability of a security force interrupting an intruder (not necessarily resulting in capture, just spotting and beginning the operation of capture and material recovery).

A computer program such as "EASI" allows one to create specific scenarios, to experiment with various responses, and to evaluate action taken in a given situation. This modeling technique is an invaluable aid to security systems, not only because it offers an inexpensive, safe alternative to a perceived intrusion, but also because within its many possible scenarios dwells the basis for nuclear safeguard systems.

It has been implied throughout the text of this module, and is not explicitly emphasized, that an efficient safeguard security system is a combination of many different fields of expertise. In the exercises that follow, the broad objectives of nuclear security should be foremost in the mind of the student: to deter, delay, and prevent acts of sabotage, theft, or diversion within a nuclear facility. With these objectives in mind and the statistical package available in this module, one has the opportunity to use some of the mechanisms employed in protecting nuclear facilities.

EXERCISES

This exercise can be performed outside of class by groups of three to four students.

For a hypothetical nuclear power plant, design a safeguard system that produces greatest possible delay for the intruder and maximizes probability of detection. (There are no definite answers.)

MATERIALS

Laboratory 1

1 programmable calculator with scientific/mathematical notation (TI Programmable 59 or SR-52 or equivalent).

Laboratory 2

8 single-channel gamma counters.

The following sealed sources of ^{137}Cs or ^{60}Co (must be a gamma-emitting isotope) should be made available and coded:

Coded Numbers	Number of Sources	Count Rate (dpm) (should be fabricated carefully for consistency)	Accountability Representation
1A, 1B, 1C, 1D	4	1,500 ea	Beginning inventory
2A, 2B, 2C, 2D	4	500 ea	Additions
3A, 3B, 3C, 4D	4	300 ea	

4A, 4B, 4C, 4D	4	200 ea	Disbursements
5A, 5B, 5C, 5D	4	400 ea	
6A, 6B, 6C, 6D	4	700 ea	
7	1	100	Losses
8A, 8B, 8C, 8D	4	100 ea	Waste
9A, 9B, 9C, 9D	4	800 ea	Ending Inventory

PROCEDURES

LABORATORY 1: EVALUATION OF PHYSICAL SECURITY BY USE OF THE "EASI" COMPUTER PROGRAM

1. Place the following "EASI" program into the computer:

EASI PROGRAM LISTING FOR THE SR-52.

Display		Key	Display		Key	Display		Key
00	46	*LBL	37	46	*LBL	74	53	(
01	15	E	38	13	C	75	42	STO
02	47	*CMS	39	65	X	76	00	0
03	25	CLR	40	43	RCL	77	06	6
04	81	HLT	41	00	0	78	55	÷
05	46	*LBL	42	03	3	79	53	(
06	11	A	43	95	=	80	01	1
07	44	SUM	44	42	STO	81	85	+
08	00	0	45	00	0	82	43	RCL
09	01	1	46	04	4	83	00	0
10	81	HLT	47	43	RCL	84	06	6
11	46	*LBL	48	00	0	85	54	)
12	12	B	49	07	7	86	54	)
13	40	*x^2	50	85	+	87	65	X
14	44	SUM	51	43	RCL	88	43	RCL
15	00	0	52	00	0	89	00	0
16	02	2	53	02	2	90	04	4
17	81	HLT	54	95	=	91	95	=
18	46	*LBL	55	30	*$\sqrt{x}$	92	85	+
19	18	*C'	56	20	*1/x	93	43	RCL
20	42	STO	57	65	X	94	00	0
21	00	0	58	53	(	95	08	8
22	03	3	59	43	RCL	96	65	X
23	81	HLT	60	00	0	97	53	-(
24	46	*LBL	61	01	1	98	01	1
25	16	*A'	62	75	—	99	75	—
26	42	STO	63	43	RCL	100	43	RCL
27	00	0	64	00	0	101	00	0
28	05	5	65	05	5	102	04	4
29	81	HLT	66	54	)	103	54	)
30	46	*LBL	67	65	X	104	95	=
31	17	*B'	68	01	1	105	42	STO
32	40	*x^2	69	93	.	106	00	0
33	42	STO	70	07	7	107	08	8
34	00	0	71	95	=	108	81	HLT
35	07	7	72	22	INV			
36	81	HLT	73	23	lnx			

2. Clear old data from memories – Press E (0 displayed).
3. An example is provided below:
    1. Press 4, 2nd, A^1 (display 4)
    2. Press 0, ., 1, 6, 2nd, B^1 (display 0.0256)
    3. Press 0, ., 9, 2nd, C^1 (display 0.9)
    4. Press 0, ., 1, A (display 0.1)
    5. Press 0, ., 0, 2, B (display 0.0004)
    6. Press 0, ., 9, 7, C (display 1.2131 ... -16)
    7. Press 3, A (display 3)
    8. Press 0, ., 3, B (display 0.09)
    9. Press 0, ., 5, A (display 0.5)
    10. Press 0, ., 1, B (display 0.01)
    11. Press 0, 1, 9, 7, C (display 0.1120..)
    12. Press 3, A (display 3)
    13. Press 0, ., 3, B (display 0.09)
    14. Press 1, ., 0, A (display 1)
    15. press 0, ., 2, B (display 0.04)
    16. Press 0, ., 3, C (display 0.35179...)

For calculations –

1. Enter E(tr), press $*A^1$ [display E(tr)]
2. Enter σ(tr), press $*B^1$ [display $\sigma(tr)^2$]
3. Enter P(c), press $*C^1$ [display P(c)]

For the intruder sequence, repeat steps E1 through E3 as many times as required.

1. Enter E(ti-last one first), press A [display E(ti)]
2. Enter σ(ti), press B [display $\sigma(ti)^2$]
3. If a P(d) is available, enter P(d), press C (display calculations)

When step-by-step input is complete, the final number is the probability of interruption, P(I).

4. Data for potential intruder sequences follows.

DATA FOR POTENTIAL INTRUDER SEQUENCES

Session A – Given the following data for potential intruder sequences, select one of each set for the total facility follow-through scenario (i.e., 1a, 2b, 3a, 4c, 5a, 6b, ...) and calculate the probability for interruption of an intruder, P(I). (Calculation begins with the last set and progresses to the first.) The smallest value of P(I) is the easiest way in for the intruder, and the student's objective is to find the easiest way in. Each set (1, 2, 3, ...) has an easiest way in, and that sequence (1a, 2b, ... or whatever) must be upgraded first.

Enter:
$E(tr) = 20.0$
$\sigma(tr) = 5.0$
$P(c) = 0.95$

Set 1. Penetration of the outer double fence barrier. Select one of three:

a. Crash the gate with a vehicle.
$E(t1) = 0.5$
$\sigma(t1) = 0.05$
$P(d) = 1.0$

b. Cut the fences.
$E(t1) = 4.0$
$\sigma(t1) = 0.5$
$P(d) = 0.5$

c. Vault the fences. (Note: if 1c is chosen, either 9b or 9c must be chosen.)
$E(t1) = 0.5$
$\sigma(t1) = 0.05$
$P(d) = 0.25$

Set 2. Cross the open area. Select one of two:

a. Avoid observed surveillance.
$E(t2) = 5.0$
$\sigma(t2) = 2.5$
$P(d) = 0.25$

b. Run flat-out.
$E(t2) = 1.0$
$\sigma(t2) = 0.2$
$P(d) = 0.65$

Set 3. Open locked door. There is only one option.
$E(t3) = 4.0$
$\sigma(t3) = 4.0$
$P(d) = 0.9$

Set 4. Travel within the building. Select one of three:

a. Carefully.
$E(t4) = 4.0$
$\sigma(t4) = 1.0$
$P(d) = 0.25$

b. Fast.
$E(t4) = 2.0$
$\sigma(t4) = 1.0$
$P(d) = 0.5$

c. With systematic search (to maximize gain).
$E(t4) = 6.0$
$\sigma(t4) = 0.5$
$P(d) = 0.1$

Set 5. Open inner door. There is only one option.
$E(t5) = 0.5$
$\sigma(t5) = 0.05$
No value for $P(d)$

Set 6. Get material, return, and open door. Select one of five:

a. Careful selection of material with radiation scan for personal safety.
$E(t6) = 15.0$
$\sigma(t6) = 5.0$
$P(d) = 1.0$

b. Careful selection of material only.
$E(t6) = 10.0$
$\sigma(t6) = 5.0$
$P(d) = 1.0$

c. Grab many cans of material and run.
$E(t6) = 5.0$
$\sigma(t6) = 2.0$
$P(d) = 1.0$

d. Grab some cans and run.
$E(t6) = 2.5$
$\sigma(t6) = 1.5$
$P(d) = 1.0$

e. Grab one can and run.
$E(t6) = 1.5$
$\sigma(t6) = 1.0$
$P(d) = 1.0$

Set 7. Go back through building and out open door. There is only one option.
$E(t7) = 6.0$
$\sigma(t7) = 3.0$
$P(d) = 0.25$

Set 8. Cross open area. Select one of two:

a. Avoid observed surveillance.
$E(t8) = 8.0$
$\sigma(t8) = 2.0$
$P(d) = 0.25$

b. As fast as possible.
$E(t8) = 2.0$
$\sigma(t8) = 1.5$
$P(d) = 0.80$

Set 9. Back through double-fenced barrier. Pick one of three (Note: if 1c was selected, 9b or 9c must be selected; if 1a or 1b was selected, 9c may not be selected):

a.	Go through existing penetration.	$E(t9) = 2.0$ $\sigma(t9) = 0.5$ $P(d) = 1.0$
b.	New penetration on other side of site.	$E(t9) = 6.0$ $\sigma(t9) = 0.5$ $P(d) = 0.05$
c.	Throw material over each fence; vault back over fences.	$E(t9) = 8.0$ $\sigma(t9) = 2.0$ $P(d) = 0.25$
Set 10.	Escape site. There is only one option.	$E(t10) = 8.0$ $\sigma(t10) = 1.0$ No value for $P(d)$

Session B – Given the following information for safeguards protection and using the sequence from Session A above, develop the best possible sequences for blocking an intruder. $E(tr)$, $\sigma(tr)$, $P(c)$, $E(ti)$, or $\sigma(ti)$ may not be changed. Only the values for $P(d)$ for each part of each set (i.e., 1a, 1b, ... 7, 8a, ...) may be changed by proper and prudent selection of highest values for $P(I)$ (or the probability of intruder detection). The objective is to make $P(I)$ as large a value as possible. Note: Returning the way the intruder went is assumed to require a second set of intruder alarms, since the first set already has alarmed. In actuality, this is not quite correct because some alarms would not "re"-function, although some would.

Device #1 has a $P(d)$ of 0.3, and 6 are available. They may be grouped or not; for example, four in Set/Part 1a and one each in 8a or 9b.

Device #2 has a P(d) of 0.9, and 4 are available.
They may be grouped.
Device #3 has a P(d) of 0.25, and 10 are available.
They may be grouped.
Device #4 has a P(d) of 1.0, and only 1 is available.
Device #5 has a P(d) of 0.6, and 6 are available.
They may be grouped.
Device #6 has a P(d) of 0.75, and 4 are available.
They may be grouped.

Assign the devices to any square available in Figure 24-6-2. Note: the number and assortments of devices available represent real conditions: A high value of P(d) indicates a very good but expensive instrument, so few are available; and, overall, not enough money is available to cover a facility as completely as it should be covered.

	a	b	c	d	e
1	1-#4	1-#1 3-#2 2-#5	4-#5	☒	☒
2	5-#3	1-#1 1-#2	☒	☒	☒
3	5-#3	☒	☒	☒	☒
4	1-#6	1-#1	none	☒	☒
6	1-#6	1-#1	none	none	none
7	1-#6	☒	☒	☒	☒
8	1-#6	1-#1	☒	☒	☒
9	none	1-#1	none	☒	☒

Figure 24-6-2. Assignment of devices.

For example, the P(d) for 4c, 6c, 6d, 6e, 9a, and 9c are 0.0 since no instruments are in those boxes. Set 1b has one of Device #1, three of Device #2, and two of Device #5. See below for sample calculations:

Grouping of P(d)s yields a new value called P'(d):

$$P'(d) = 1 - [(1 - P(d))^n]$$

where:

n = number of the same device.

For four of Device #1, $P'(d) = 1 - (1 - 0.3)^4 = 1 - (0.7)^4 = 0.7599$.

For the example given in Box 1b of Figure 24-6-2, the following calculations are required:

The P(d) for device #1 is 0.3; since there is only one — #1, $P'(d) = P(d) - 0.3$.

The P'(d) for three of Device #2 - $1 = (1 - 0.9)^3 = 1 - (0.1)^3 = 0.999$.

The P'(d) for two of Device #5 = $1 - (1 - 0.6)^2 = 1 - (0.4)^2 = 0.84$.

Now calculate P''(d) for the entire Box:

$$P''(d) = 1 - (1-P'(d)_1)(1-P'(d)_2) \ldots$$

Note: P''(d) = P(d) if there are no P'(d)s and only one P(d).
or = P'(d) if there is only one P'(d) and no P(d)s.

For the above:

$$\begin{aligned} P''(d) &= 1 - (1 - 0.3)(1 - 0.999)(1 - 0.84) \\ &= 1 - (0.7)(0.001)(0.16) \\ &= 0.999888. \end{aligned}$$

Therefore, P''(d) for Set/Part 1b is 0.999888, and now call 0.999888 the overall P(d) for 1b.

After calculating all P(d)s, check each set (i.e., 1, 2, 3, ... etc.) as follows:

a. Press E.
b. Set *A', *B', *C' (i.e., enter the appropriate numbers).
c. Enter appropriate numbers for A, B, and C.
d. Write down the numbers for each set/part (including those with a P(d) of 0.0).
e. The lowest value of P(d) for each set is the worst design. Note that one down for the sequence.
f. List worst sequence (1a, 2b, ...).
g. Now calculate the entire P(I).

Answer the following questions from the overall experiment (Sessions A and B) and from everyone's results:

a. Where did the greatest value of P(I) physically come from? That is, from what set did it come? How does this influence timing for intruder detection?
b. What is the most important factor in the interruption of an intruder?

LABORATORY 2: DEVELOPMENT AND BALANCING OF AN INVENTORY OF GAMMA SOURCES

1. Divide the class into groups of 2, 3, or 4.
2. Give each group a set (1, 2, 3, 4, 5, 6, 7, 8, or 9) of the gamma-emitting sources. Each individual should have a separate counter.

3. Count a background on the gamma counters for 30 minutes. The background value is B: b B , and the background value is B b. Divide by 30 minutes: B/30 b/30 B' b'.
4. Count each individual sample (1A, 3C, etc.) for 10 minutes.
    a. Make certain that all of a set are counted on the same counter in order that the same background and counter peculiarities are localized within the same set.
    b. Number 7 can be counted on any counter, and that counter's background is used subsequently.
    c. There will be 33 sample counts for 1A, 1B, 1C, 1D, 2A, 2B, ... 9C, and 9D.
    d. Let any one count be X.

Calculate x: $x = \sqrt{X}$, thus one has X ± x.
Divide by ten minutes: $X/10 \pm x/10 = X' \pm x$.
Now subtract background: $(X' - B') = Y'$ and $y' = \sqrt{b'^2 + x'^2}$.
Thus, each sample count has been reduced to a final value of $Y' \pm y'$.

Now, for each set (except Set 7) – say, for Set 4 –

$$\frac{Y'_A + Y'_B + Y'_C + Y'_D}{4} = Z.$$

(Note: For Set 7, $Y' = Z$.)

$$\sqrt{y'^2_A + y'^2_B + y'^2_C + y'^2_D} = z.$$

(Note: for Set 7, $y' = z$.)

    e. For any given set, the values as determined for the inventory are given by Z z, where Z is the set value and z is the uncertainty on Z.

5. Take all determined set data and calculate an inventory. Calculate the following:
    a. Book inventory from Sets 1 through 8.
    b. Physical inventory, which is Set 9.
    c. MUF.
    d. LEMUF.
    e. Should a diversion be suspected?
    f. Repeat the entire counting/calculational procedure a second and third time, and answer parts a through e for each time.

DATA TABLE

None.

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1. Match the following:

________ Pu.

________ Outer perimeter barrier.

________ Isolation.

________ Intrusion alarm.

________ LEMUF.

________ MUF.

________ Equipment barrier.

________ Uncertainty on MUF.

________ Vital equipment.

________ Fissile waste.

A. One sigma (1σ).

B. Two sigma (2σ).

C. Security equipment.

D. Material that cannot be found.

E. SNM evaporator.

F. Zone.

G. Fence.

H. SSNM.

I. Concrete cell wall.

J. Material losses.

2. The probability for an intruder being detected by an instrument is P. (1-P) is the probability for the intruder not being detected. The probability of an intruder not being detected by 2, 3, or more of the same instrument is $(1-P)^2$, $(1-P)^3$, or $(1-P)^{\text{more}}$. The probability of an intruder being detected is $1-(1-P)^2$, $1-(1-P)^3$, or $1-(1-P)^{\text{more}}$. Likewise, for four different instruments the probability is –

$(1-P_1)(1-P_2)(1-P_3)(1-P_4)$ (not being detected) and $1-[(1-P_1)(1-P_2)(1-P_3)(1-P_4)]$ (being detected).

Answer questions a through d, given the following values:

$P_1 = 0.3$; cost of #1 is \$300.
$P_2 = 0.6$; cost of #2 is \$400.
$P_3 = 0.9$; cost of #3 is \$14,000.
$P_4 = 0.2$; cost of #4 is \$150.

a. Which of the following combinations is the best, and why?

  i. One of #1 and two of #4.
  ii. Two of #1.
  iii. Six of #4.
  iv. One of #1 and one of #2.
  v. Two of #2.
  vi. Two of #1 and one of #4.

(Hint: Calculate the probability of an intruder being caught for each combination.)

Combination	Probability		Cost
	Not Catching	Catching	
i	0.45	0.55	600
ii	0.49	0.51	600
iii	0.26	0.74	900
iv	0.28	0.72	700
v	0.16	0.84	800
vi	0.39	0.61	750

b. Only $750 is available for alarming devices. What is the best solution in question 2a above? Why?

c. If there is one each of #1, #2, #3, or #4, what is the probability for detecting an intruder?

d. Given the following data, (1) Which is a better detector for the money? (2) Which, of #3 or #4, is a better detector for the money? (3) Which is the best of the four detectors? (Hint: Only compare #2 and #4.)

	Unit #1		Unit #2	
# of Units	Probability of Detection	Cost	Probability of Detection	Cost
1	0.30	$300	0.60	$400
2	0.51	$600	0.84	$800
3	0.66	$900	0.94	$1200

	Unit #3		Unit #4	
# of Units	Probability of Detection	Cost	Probability of Detection	Cost
1	0.90	$14,000	0.20	$ 150
2	0.99	$28,000	0.36	$ 300
3	--	--	0.49	$ 450
4	--	--	0.59	$ 600
5	--	--	0.67	$ 750
8	--	--	0.83	$1200
10	--	--	0.89	$1500

e. How many of each type of device are required for at least 95% detection confidence? For example, the probability of two #2s is 0.84 or less than 0.95: more than two are required. (Hint: 95% detection confidence is 5% (or 0.05) of not detecting the intruder.)

(1) Number of #1 units –

(2) Number of #2 units –

(3) Number of #3 units –

(4) Number of #4 units –

3. True or False

_______ Pu is a source material.

_______ Th is a source material.

_______ SSNM is a fissile material.

_______ Enrichment is the percentage of O in UO_2.

_______ Safeguards are concerned only with nuclear materials.

_______ Safety is unimportant when compared to safeguards.

_______ Explosives cannot be used in nuclear facilities.

_______ Accidents never happen in nuclear facilities.

_______ An intruder can never be a plant employee.

_______ The Federal government requires inventories.

_______ A goal of any safeguards program is to prevent sabotage.

_______ The safeguards system is most effective if the security forces continually test it.

_______ A false alarm does not need to be investigated.

_______ An employee has no obligation to "rat" on another employee.

_______ Lunchboxes must be searched on entry into a safeguarded site.

_______ An employee who loses his or her picture badge has no requirement to notify plant management immediately.

_______ MUF is the uncertainty on the book inventory/physical inventory difference.

_______ LEMUF is the uncertainty on the book inventory/physical inventory difference.

_______ Of the isotopes of ^{233}U, ^{234}U, ^{235}U, and ^{238}U, only two – ^{233}U and ^{235}U – are considered SSNM.

4. Multiple Choice. Pick one answer; write A if all are correct; or write N if none are correct.

   a. Enrichment level determines the difference between SSNM and SNM. That difference is at what percentage enrichment?
      (1) 10%.
      (2) 20%.
      (3) 50%.
      (4) 93.5%.
      (5) 100%.

   b. Radiation comes from radioactive decay. Of the following nuclear fuel elements, which one is (ones are) radioactive?
      (1) Th.
      (2) U.
      (3) Pu.

   c. Which of these contain no fissile isotope?
      (1) Th.
      (2) U.
      (3) Pu.

d. A MUF of greater value than LEMUF could mean –
   (1) A diversion.
   (2) A theft.
   (3) An undetected tank leak.
   (4) An inventory error.
   (5) A security problem.

e. The site boundary could be made into a site barrier with –
   (1) A vehicle trap and deep ditch.
   (2) A wooden fence.
   (3) Reinforced gates.
   (4) Several armed guard posts with total area observation.
   (5) Multiple intrusion alarms.

f. An employee who works in a nuclear power plant must carry –
   (1) A picture badge, radiation monitor, and an extra change of clothes.
   (2) A picture badge, a radiation monitor, and a lunchbox.
   (3) A picture badge, a radiation monitor, and a handbag.
   (4) A change of clothes, a lunchbox, and a handbag.
   (5) Some of the above individual articles (such as a lunchbox) – at least two, but not all.

g. Identification of an employee at the guarded gate is uncertain. The identification can be made more certain by all of the following except –
   (1) The employee's signature.
   (2) His/her driver's license, although it has no picture.

(3) His/her buddy's word.
(4) A voice print although he/she has a cold.
(5) Employee's height, weight, and other physical description.

h. All of the following are true (correct) about a safeguarded plant except one:
(1) All employees individually can improve safeguardability by obeying all plant rules.
(2) Safeguards can be penetrated, one way or the other.
(3) Security guards are more capable and versatile than electronics instrumentation.
(4) A safeguarded facility, if shut down and without non-security employees, still must have security guards on site and/or portal.
(5) Plant management makes all of the security rules for a facility.

i. SSNM cannot be kept in –
(1) Operations buildings.
(2) Storage vaults.
(3) Lunchrooms.
(4) Process vessels.
(5) Laboratories.

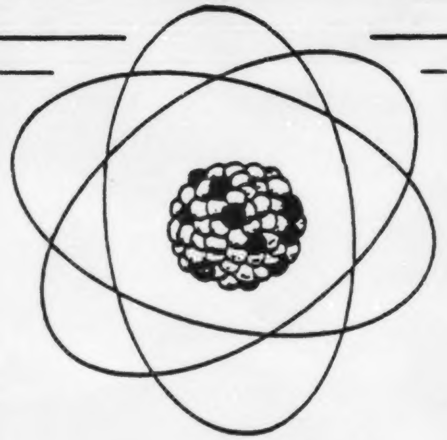
j. A safeguarded facility could have –
(1) No SNM.
(2) Corrosive acids.
(3) Flammable organic liquids.
(4) Lunchrooms.
(5) Heavy moving equipment.

5. An inventory balance shows –

	Pu (kg)	^{233}U (kg)
Beginning inventory	27.629	175,968
Disbursements	75.962	150.420
Additions	49.555	3.762
Losses	0.057	0.476

a. Calculate book inventory for Pu and ^{233}U. Ignore uncertainties.

b. All entries for Pu have a ± 1% uncertainty. Calculate the uncertainty on the Pu book inventory.



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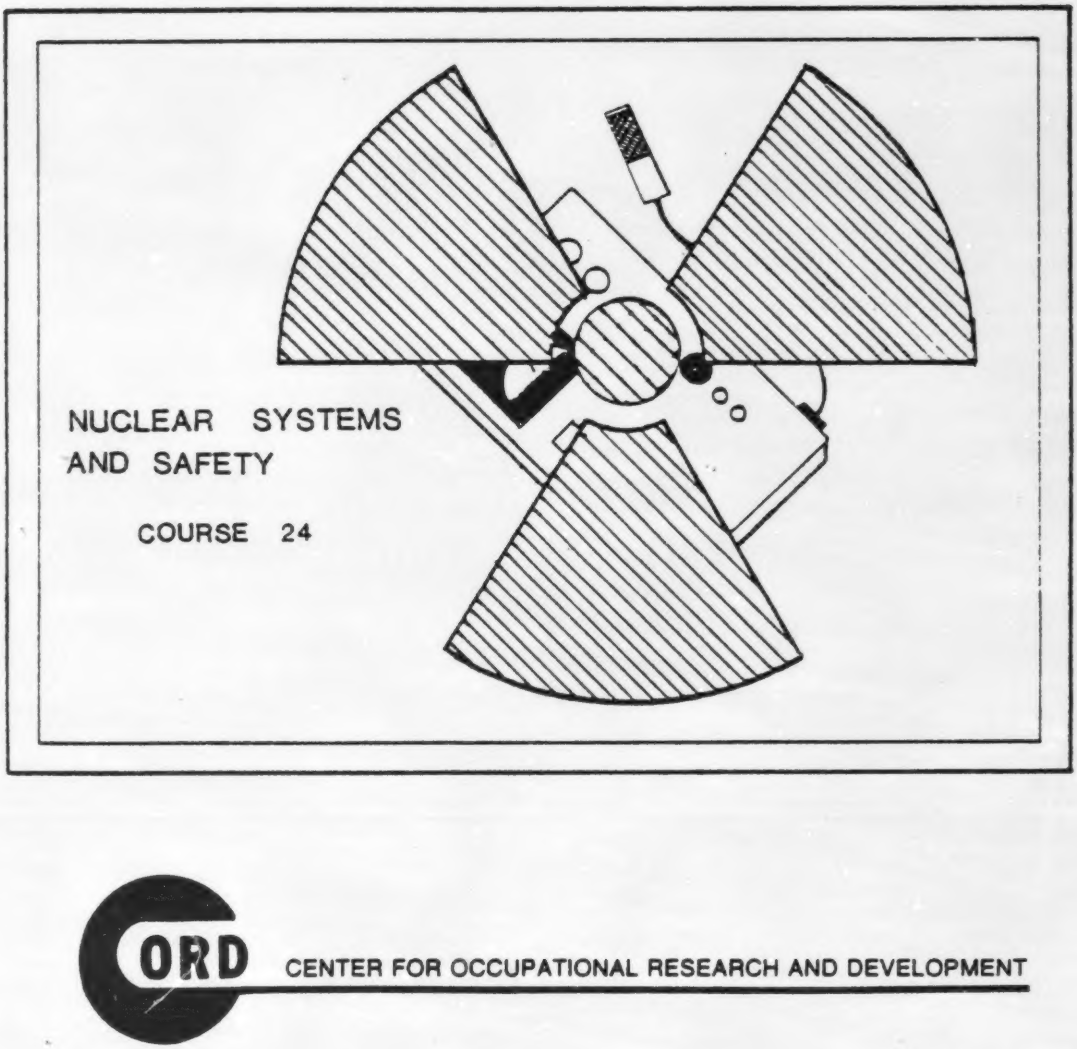
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for
COURSE 24, NUCLEAR SYSTEMS AND SAFETY (SIX MODULES)

NOTE: Each module in this Course is numbered separately, and the page numbers begin with page 1 in each module. Therefore, this Index lists for each entry the module number (underlined), followed by a comma and the page number. Figures and tables are listed by their numbers, followed by the module and page number in parentheses. A Master Index is available in a separate volume. This general index merges all index entries from the 35 Nuclear Courses. It is a valuable research tool, with which the student or researcher can locate all the references to any subject within the 35 Courses.

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